6T-2700131

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

July 31, 2001

MEMORANDUM TO: ACRS Members

FROM: Michael T. Markley, Senior Staff Engineer

SUBJECT: ASME STANDARD FOR PRA FOR NUCLEAR POWER PLANT APPLICATIONS AND SUPPORTING WHITE PAPER

The purpose of this memorandum is to forward Revision 14A of the ASME Standard for PRA for Nuclear Power Plant Applications and the supporting White Paper for consideration by the Committee.

Background

The ACRS previously offered comments on the proposed ASME Standard (internal events) in letters dated March 25, 1999 (Revision 10) and July 20, 2000 (Revision 12). The ASME Project Team has incorporated stakeholder comments and the proposed Standard has been issued for a last round of public comments before final approval by ASME.

The ACRS staff suggested and Dr. Apostolakis agrees that there is no need for the Committee to review Revision 14A of the ASME Standard. It is expected that the NRC staff will consider endorsing the final version of the ASME Standard in a future revision to Regulatory Guide 1.174 with exceptions and clarifications, as appropriate.

In parallel, it will be important for the Committee to consider the relationship between the ASME Standard and other ongoing initiatives related to PRA quality, e.g., ANS Standards for external events and for low-power and shutdown operations, industry peer review certification guidelines (NEI 00-02), and proposed Revision 1 (Draft Guide 1110) to Regulatory Guide 1.174 (SECY-00-0162).

Expected Committee Action

No Committee action is expected at this time. Revision 14A to the ASME Standard is provided for information only. The Committee plans to review a future revision to Regulatory Guide 1.174 endorsing the ASME Standard during a future meeting.

Attachments: ACRS letters 3/25/99 and 7/20/00
ASME Standard Rev. 14A and associated White Paper

cc w/o attach: ACRS Staff and Fellows

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

July 20, 2000

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Dr. Travers:

SUBJECT: PROPOSED FINAL ASME STANDARD FOR PROBABILISTIC RISK ASSESSMENT FOR NUCLEAR POWER PLANT APPLICATIONS

During the 474th meeting of the Advisory Committee on Reactor Safeguards, July 12-14, 2000, we met with representatives of the American Society of Mechanical Engineers (ASME) Committee on Nuclear Risk Management (CNRM) to discuss the proposed final Standard for Probabilistic Risk Assessment (PRA) for Nuclear Power Plant Applications. Our Subcommittee on Reliability and PRA met with the ASME CNRM on June 28, 2000, to discuss this matter. We previously reviewed a draft version of the ASME Standard and commented in a letter dated March 25, 1999.

Conclusions and Recommendations

1. The proposed Standard is not a traditional "design-to" engineering standard or a procedures guide. Consequently, any argument that a PRA should be accepted by the staff simply because it meets the Standard would not be valid.

2. The Standard should be useful because it provides a framework for the systematic assessment of PRA elements. This will aid staff reviews by identifying weak elements in a PRA. Because the Standard can accommodate a wide range of PRA quality, however, the staff will still need to make a case-by-case assessment of the adequacy of the PRA.

3. The three categories of PRA requirements proposed in the Standard deal reasonably with the wide range of risk-informed decisions. The differences among the categories should be delineated more clearly, especially the treatment of uncertainties.

4. The discussion of the categories of requirements needed for particular regulatory applications that is given in Section 1.5, "Application Categories," can be misleading and should be deleted.

5. More guidance and examples should be given on the circumstances under which supplementary analyses would be needed and how they would enhance the scope and level of detail in a PRA.

Discussion

The quality of PRA is at the heart of a successful risk-informed regulatory system. The term "quality" includes many things, such as issues of scope, detail, and technical adequacy of the analyses. PRAs are very ambitious. To model everything that is relevant in a particular situation, including hardware failures, human performance, as well as physical and chemical phenomena, is extremely difficult. Defining PRA quality *a priori* is a highly subjective and very difficult task, given the varied nature of potential risk-informed decisions. Thus, PRA quality should be evaluated in the context of the decision the PRA supports. If, for instance, a particular decision is insensitive to recovery actions, a PRA that does not include such actions would not suffer in quality for that particular decision.

The Standard recognizes this difficulty and proposes three categories of requirements that determine the range of applications for which a PRA would be appropriate. The delineation of the differences among categories is not always clear and this situation is exacerbated by the fact that the Standard relies primarily on tables with limited accompanying text. More details on the differences among the categories and further elaboration on the requirements would be beneficial.

The NRC staff will ultimately have to decide whether the submitted risk information is sufficient and of adequate quality to support a particular risk-informed decision. The categories and the associated requirements will facilitate this process by helping all parties involved establish a common PRA language and by providing a framework within which potential weaknesses of the PRA could be identified early in the decisionmaking process.

The Standard should not be viewed in the same way as other, more traditional, "design-to" standards usually associated with ASME. PRAs of a wide range of quality could be said to meet the requirements of the Standard. Consequently, any argument that a PRA should be accepted by the staff simply because it meets the Standard is moot. The discussion of the categories of requirements needed for a particular regulatory application provided in Section 1.5 of the Standard should be deleted to avoid misunderstandings and misleading expectations. We were told by the ASME representatives that they would consider revising this Section to avoid these problems.

For a given application, the Standard allows the use of supplementary analyses to augment the PRA but does not provide guidance on the scope and level of detail of these analyses relative to that provided for the categories. Lack of such guidance may increase the NRC staff effort required to assess the appropriateness of the supplementary analyses in risk-informed decisionmaking.

We offered a number of detailed comments on the Standard that the ASME representatives agreed to consider. We look forward to reviewing the staff's work related to this matter.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Letter dated June 14, 2000, from G. M. Eisenberg, ASME International, to M. Markley, ACRS, transmitting Draft #12 of Proposed ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, dated May 30, 2000.
2. American Society of Mechanical Engineers, "White Paper and Guidance to Reviewers of the Draft ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," dated June 13, 2000.
3. Letter dated March 25, 1999, from Dana A. Powers, Chairman, Advisory Committee on Reactor Safeguards, to William D. Travers, Executive Director for Operations, NRC, Subject: Proposed ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications (Phase 1).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

March 25, 1999

Dr. William D. Travers
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Dr. Travers:

SUBJECT: PROPOSED ASME STANDARD FOR PROBABILISTIC RISK ASSESSMENT FOR NUCLEAR POWER PLANT APPLICATIONS (PHASE 1)

During the 460th meeting of the Advisory Committee on Reactor Safeguards, March 10-13, 1999, we met with representatives of the American Society of Mechanical Engineers (ASME) Committee on Nuclear Risk Management (CNRM) to discuss the proposed Standard for Probabilistic Risk Assessment (PRA) for Nuclear Power Plant Applications (Phase 1). The purpose of this Standard is to provide a means to ensure that the technical quality of PRAs is sufficient to support the regulatory review and approval of licensee risk-informed applications. We also had the benefit of the documents referenced.

Conclusions and Recommendations

1. The proposed Standard has the potential of being very useful to both the industry and the NRC. Although additional work remains, the overall approach to defining necessary PRA requirements is good.

2. Subsection 3.5 on the use of expert judgment and the associated nonmandatory guidance in Appendix A are inconsistent with other parts of the Standard and should be revised. Subsection 3.5 should identify the major issues involving the use of expert opinion in a PRA and not focus on a particular approach.

3. We agree with the CNRM decision to move Section 7 to the beginning of the Standard to present the risk assessment application process early in the document.

4. Consideration should be given in the Standard to recommending participatory peer review throughout the development or application of the PRA in preference to *a posteriori* review.

Discussion

The move toward a risk-informed regulatory system has increased awareness of the need to examine the quality of PRA methodologies. Risk information used for regulatory decisions must be based on credible models and methods.

The lack of confidence in the quality of PRAs will impede their use in the regulatory process. For example, the Individual Plant Examination (IPE) Insights Report (NUREG-1560) showed that there is variability in PRA results that can be attributed to different analytical tools used by licensees. On the basis of its review of licensee IPEs, the staff determined that assumptions used by some licensees were unacceptable and requested those licensees to improve their analyses. The development of a Standard that defines the necessary and minimum requirements for acceptable PRA quality is, therefore, essential.

Developing this Standard is not a straightforward process. If the Standard is too prescriptive, it could impede the further development and refinement of PRA models. On the other hand, simply listing all the methods and models that analysts have used or proposed in the past is not helpful because it presents all such tools as being equally credible or useful when, in fact, experience has shown that they are not.

We believe that the CNRM, who developed the proposed Standard, has established an appropriate balance between specificity and flexibility. The proposed Standard provides requirements that the CNRM believes are necessary for a quality PRA. Although there are references to methods in which there is broad consensus on their appropriateness, the CNRM has wisely refrained from being overly prescriptive in areas where the choice of methods is less clear. Because the actual methods for satisfying the requirements are not prescribed, merely meeting the requirements does not guarantee that a PRA will be of acceptable quality. Thus, the Standard also requires a peer review process to ensure acceptable quality. We agree with the CNRM that a robust peer review process is at present the best way to assess quality. Consideration should be given in the Standard to recommending participatory peer review throughout the development or application of the PRA in preference to just a review after completion of the work.

An exception to the CNRM decision not to specify methods is the treatment of expert judgment. Expert judgment has proven to be a ubiquitous element of modern PRAs for nuclear power plants. Overall, the proposed treatment of expert judgment in the Standard and in the nonmandatory Appendix A touches on nearly all the points that are needed. It puts an unwarranted emphasis on a particular approach to expert judgment. Subsection 3.5 should be revised to be consistent with the remainder of the Standard. Also, since it is not common practice to employ formal expert judgment methods in Level 1 PRAs, a discussion of the conditions requiring such treatment, with examples, would be very useful.

Subsection 7.5 requires that the users determine whether the scope and level of detail of the Standard are sufficient for an application and to provide a technical basis for this determination. Additional guidance should be provided in the Standard to clarify what is expected of the users.

To date, the work done to develop the proposed Standard and associated guidance is commendable. The Standard, when integrated with other industry and NRC initiatives, should greatly enhance progress toward risk-informed nuclear operations and regulatory decisionmaking. We applaud the staff for initiating this effort and for actively participating in the working committees.

We offer detailed comments in the attachment to this letter for the benefit of the CNRM in developing the proposed final version of the Standard and the NRC staff in considering possible endorsement. We look forward to reviewing the proposed final Standard following the reconciliation of public comments.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. American Society of Mechanical Engineers, ASME RA-S-1999 Edition Draft #10, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," draft released for public comment, dated February 1, 1999.
2. American Society of Mechanical Engineers, "White Paper and Guidance for Reviewers of the Draft ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," received February 8, 1999.
3. U.S. Nuclear Regulatory Commission, NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," Vols. 1-3, December 1997.
4. U.S. Nuclear Regulatory Commission, NUREG-1150, "Severe Accident Risks - An Assessment for Five U.S. Nuclear Power Plants," December 1990.
5. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.

Attachment: As Stated

ATTACHMENT

Detailed Comments on Proposed ASME Standard for PRA for Nuclear Power Plant Applications (Phase1)

1.1 Scope

Subsection 1.1 states that the Standard sets forth criteria and methods for developing and applying PRA. It should be made clear that the emphasis is on criteria and that particular methods are not prescribed.

2. DEFINITIONS

A. Section 2 requires a thorough review. Considering the broad range of potential applications for this Standard, close scrutiny should be given to ensuring that the definitions are consistent with generally accepted reactor and risk terminology and that terminology used in each section of the Standard is appropriately addressed.

B. Many of the listed definitions are not needed. For example, there is no need to describe a mathematical method such as Monte Carlo simulation. Similarly, there is no need to define a "severe accident." The inclusion of the words "beyond design basis" in the definition is not appropriate.

C. Some of the listed definitions are not useful. For example, an "importance measure" is called a mathematical expression that defines a quantity of interest.

D. Several of the listed definitions are inaccurate or incorrect. Examples of the former are the definitions of "station blackout," "core damage frequency," "unavailability," and "cut sets." An example of the latter is the definition of the "failure rate."

E. Many terms in the text, which should be included in the definitions, are not defined in Section 2. Examples are: EOPs, I&C, ECCS, safety-related SSCs, aleatory and epistemic uncertainties, and single-failure criterion.

3.1 Scope

"Internal Flooding Analysis" is located in the wrong place in Fig. 3.1-1, "Technical Elements of a PRA Model."

3.2 Plant Familiarization

Page 18: An important example of the plant familiarization that should be made explicit is crew performance on simulators during known, generic, time-critical sequences. This provides an appropriate understanding of man-machine interaction.

3.3.1 Initiating Event Analysis

A list of the initiating events that have been used in PRAs should be included with appropriate guidance.

3.3.2 Sequence Development

The explicit description of conditional split fractions and of fault tree linking is appropriate because they are established and accepted approaches. Similarly, a portion of the discussion on event sequence diagrams and system dependency matrices should be removed from the nonmandatory Appendix A and relocated into the main body of the Standard.

3.3.3 Success Criteria

A. Page 23: The list of high-level functions should also include neutronic shutdown.

B. Page 23: Criteria resulting from neutronic analyses should be added to the list of requirements.

C. Page 23: The statement that bounding analyses can be used conflicts with Subparagraph 3.3.4.3, "Use of Realistic Success Criteria."

D. Page 23: Second column: specifies that "Bounding thermal-hydraulic analyses from the plant's SAR ... may be used when detailed analyses are not practical." This statement conflicts with the word "shall" used in Subparagraph 3.3.4.3 to ensure that realistic criteria are used.

3.3.4 Systems Analysis

A. The Standard should caution users that the calculation of the average unavailability of systems with redundant trains is not the product of the average unavailabilities of the individual trains. The time-averaging process introduces dependencies among train unavailabilities.

B. Page 32: The definition of the term "common-cause equipment failure" is not consistent with the definition provided in Section 2.

3.3.5 Data Analysis

A. Page 35: Although it is stated that the subjectivist approach to probability ought to be adopted, the Standard proceeds to discuss frequentist methods (Subparagraphs 3.3.5.1.4 and 3.3.5.3.5) that are inconsistent with this recommendation on the subjectivist approach.

B. Page 35: The Standard should be clarified to state when frequen. can be used and for what purpose. It should state that no PRA that. uncertainty analysis has considered these methods useful.

C. Page 40: The Standard should be clarified to state that the analysis of comm. cause failures will require the use of generic data that are applicable to the specific plant under analysis.

3.3.6 Human Reliability Analysis

Page 45: The statement in Subparagraph 3.3.6.3.1 that recovery actions shall be limited to those actions for which some procedural guidance is provided or for which operators receive frequent training is inconsistent with the statement in 3.3.7.6 that extraordinary recovery actions that are not proceduralized shall be justified in the analysis.

3.3.8 Level 1 Quantification and Review of Results

A. Page 51: It is not clear what the CNRM means in Paragraph 3.3.8.1.2 by the exception stating, "If only point estimate quantification is completed, that point estimate shall be the mean." Does this mean that the "mean value" should be calculated using rigorous methods? What does the CNRM mean by "point estimates"?

B. Page 51: The requirement in Subparagraph 3.3.8.1.3 that model uncertainty be evaluated needs additional discussion. This evaluation can range from a quick estimate of uncertainty to the use of formal methods for expert opinion elicitation, as was done in NUREG-1150. Furthermore, additional guidance should be provided to clarify how the sensitivity studies should be done and how the results may be used.

3.3.9 Level 1 and Level 2 Interface

A. The determination of uncertainty should be given more discussion and a more prominent position in the Standard.

B. Page 55: The second example of accident sequence characteristics that should be considered refers to the "RCS pressure at core damage." This should be replaced with the "RCS pressure at the time of vessel penetration."

C. There should be a brief discussion on how to extract the Regulatory Guide 1.174 equivalent [large, early release frequency (LERF)] from the results of the detailed Level 2 PRA analysis.

3.4.2 Mapping of Level 1 Sequences

These risk assessments depend on the adequacy of the user's modeling of the physical response of the entire system to accident conditions. For example, whether or not a fan

cooler fails due to internal waterhammer, or waterhammer in a piece of pipe to which it is connected, depends on many details of the piping geometry, ups and downs, water-storage tanks, starting transients of pumps when connected to the entire system of pipes, valves, tees and components, the rate of rise of containment temperature and humidity, etc. A technical analysis, including evaluation of uncertainties in modeling, plays the biggest role in assessing failure probability, rather than some characteristics of the device itself. The PRA is fragile if it is not based on the comprehensive analysis of system response. The Standard should reflect this dependence.

3.4.4 Radionuclide Release

A. Page 62: The last bullet calls for "the size distribution of radioactive material released in the form of an aerosol." Isn't this a time-dependent parameter? Is it to be specified as a function of time or an average?

B. Table 3.4.4-1 may be overkill with respect to the needs for determining LERF. Not all of the fission products are significant for LERF although they can be for a full Level 2 PRA analysis.

C. Page 64: Calls for including the release energy in the radionuclide source term. Is this the temperature, the enthalpy, the internal energy? Does it include radioactive energy?

D. Table 3.4.4-2 does not contain all of the key uncertainties. It should be expanded.

E. Page 65: Under the first example, the comment is made that "higher retention efficiencies were attributed to sequences involving low coolant system pressure than those involving high pressure." Is this correct? Was it not the inverse?

F. There is a need to discuss the release and effects of non-radioactive aerosols from the core.

3.5 Expert Judgment

A. What are the criteria for deciding when expert judgment must not be used in order to have a PRA of acceptable quality?

B. When are higher level treatments of expert judgment necessary to ensure that a PRA of acceptable levels of quality is produced? If there are not definable occasions when higher order treatment is needed to ensure adequate quality, why does not the Standard specify the minimum acceptable level of treatment and leave to guidance (i.e., in the Appendix) the discussion of higher levels of treatment that are not likely to ever be used?

C. The Standard requires that the problem to be addressed by the experts be specified in advance. Why is it not required that the experts be allowed to modify

the problem? This is allowed in the nonmandatory guidance in Appendix A and would seem to be wise since the experts are very likely to know more about the issue than the PRA team.

D. The Standard requires that the degree of importance of the issue be determined, but provides no quantitative indication of the measure of importance. How can this be omitted if the goal is to have a PRA of adequate quality? The nonmandatory guidance provides some qualitative indications of importance that are sufficiently vague to ensure that all issues can be relegated either to the lowest or to the highest category of importance. Is it not possible to provide a specification of the measure of importance of an issue?

E. The Standard requires also that the complexity of the issue be determined. Here even the nonmandatory guidance is of no help. In the nonmandatory guidance, levels of complexity are described. In some cases these levels are described as "... levels of complexity of the issue under consideration..." (p.103-A-3.5.1[2.2]). But elsewhere these are described as "... levels of complexity in the use of experts..." (p.101-A) and it is apparent that this is the real meaning of the terms. What is the meaning of the "level of complexity of the issue" as specified in Paragraph 3.5.1(b)? What is the measure of complexity to be used?

F. Paragraph 3.5.3: The decision to use outside experts rather than relying on the collective wisdom of the PRA analysis team would seem to be a step in the direction of the quality of the PRA that may not be needed. The decision to do this is left completely to the judgment of the team. Surely, it must be known that there are issues that can be resolved properly for the purposes of producing a PRA of adequate quality only by using outside experts. Why are the characteristics of these issues not described?

G. Paragraph 3.5.4: A crucial step in the formulation of the expert judgment for the PRA is the aggregation of the various expert judgments. No requirements for this step are provided. How is this absence of any specification for such a crucial step consistent with the goal of having a PRA that has adequate quality?

H. Subparagraphs 3.5.4.1 and 3.5.4.2: Regarding Levels A, B, C, and D, there is no indication in the Standard of what these Levels are. The nonmandatory guidance provides some idea of what they are for those who choose to follow this guidance. What are the meanings of Levels A, B, C, and D for those who elect not to follow the nonmandatory guidance? People familiar with the formulation of standards should be added to the group preparing this Standard. Similar flaws arise throughout the discussion in these Subparagraphs. What are four levels of consensus? If the guidance in Appendix A is to be followed, the Standard should require it. Otherwise, revise the Standard so that it stands alone.

I. Why are requirements for documentation of the expert judgment process not mentioned by reference in Subsection 3.5?

4. Documentation

The CNRM provides a listing of specific documentation requirements for a PRA that reflects, one-for-one, the listing of Risk Assessment Technical Requirements provided in Section 3. Although this listing is redundant, a concise listing of these documentation requirements would be helpful in avoiding diverse assessments of the Section 3 requirements. A careful review of Section 4 should follow the rewrite of Section 3. Also, where documentation requirements are stated in Section 4, a more specific statement of the kind of assessments necessary to satisfy these requirements should be useful, e.g., in the evaluation of the consequences of a residual heat removal system train failure, an adequate thermal-hydraulics analysis of system response is needed.

6.2 Review Team Personnel Qualifications

A. Define or describe the requirements for "indoctrination on the PRA process."

B. How were the various experience requirements established? e.g., "The team, collectively, shall have 15 years of experience in performing the activities related to the technical elements of the nuclear power plant PRA identified in Section 3 of this Standard."

C. The last paragraph is a documentation requirement, which may not belong in Subsection 6.2.

6.5 Review of Technical Elements

Consider a generic approach to defining when detailed or limited review is required. Consider reducing the redundancy of review guidance.

7.6 Determination of Scope and Level of Detail of Standard are Sufficient for Application

We are perplexed by the suggestion in Subsection 7.5 that the users determine whether the Standard is sufficient. Subsection 7.5 should be expanded to provide detailed guidance regarding the determination that the Standard is not sufficient to support a particular application and why alternative methods are needed. Also, a new section should be added to provide guidance on how users may recommend improvements to the Standard and for ASME to maintain and update the Standard.

PREDECISIONAL DRAFT **PROVIDED FOR INTERNAL ACRS USE ONLY**

White paper and guidance for reviewers of the draft ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications

A project team under the ASME Committee on Nuclear Risk Management (CNRM) is drafting a new standard on the use of PRA to support risk-informed applications at nuclear power plants. At this point in the development of the standard, the team is seeking broad review and comment. The feedback received will be considered by the project team in revising this draft before the standard is submitted to CNRM as part of the formal ASME consensus process.

To facilitate quality reviews and reviewer understanding of the scope and content of this draft, ASME will post additional material, in the form of appendices to this draft, on the ASME website on or before February 17, 1999. Also, a public workshop will be held to give reviewers an opportunity to meet with members of the project team. Details of this workshop will be posted on the ASME web site on or before February 1, 1999.

The comment period will end on April 29, 1999. Comments should be submitted to Jess Moon, ASME staff secretary of the CNRM Project Team, at moonj@asme.org **not later than April 29, 1999.**

Background

The origin of this effort is described in the Foreword of the accompanying draft standard. The purpose of this standard is to provide a way to ensure that the technical quality of a PRA used to support a risk informed application is sufficiently adequate for that application, such that the level of regulatory review needed for approval of that application is minimized. This is done by, first, describing a "reference" PRA in terms of its technical elements (Section 3), required documentation (Section 4), configuration control (Section 5) and peer review requirements (Section 6). The standard then describes a process (Section 7) for:

- determining the extent to which the reference PRA technical elements are necessary and sufficient to support a particular risk informed application,
- comparing the plant PRA to the reference PRA, and
- evaluating the significance **to this specific application** of any differences between the reference PRA and the plant PRA.

It must be emphasized that **the "reference" PRA in this standard is meant to be used only in the context of this process.** In particular, it should not be assumed that the "reference" PRA in this standard is a model for nuclear power plant PRAs. On the contrary, the standard recognizes that existing plant PRAs, and existing methods for documentation, configuration control and peer review, will differ from the standard. The process in Section 7 allows for differences between the standard and a plant PRA, and it provides for other ways of augmenting an application when the differences are significant to that specific application. In selecting the technical elements in Section 3, for example, the project team attempted to describe a PRA model that provides a "reasonable estimation of core damage frequency." No attempt was made to quantify the robustness and value of this PRA outside its use in the above process for risk informed applications.

As explained in the General Requirements (Section 1), this draft of the standard is limited to Level 1 analysis of internal events while the plant is at full power, and it excludes internal fires and external events. In addition, it includes a limited Level 2 analysis sufficient to evaluate the large early release frequency (LERF). It is the intent of the ASME Committee on Nuclear Risk Management to expand this scope (e.g., to a complete Level 2 analysis) in later versions of this standard.

The following are specific areas where the project team would appreciate feedback from reviewers.

General requirements

Are the scope (Section 1.1) and applicability (Section 1.2) clear? Are they consistent with the background information provided above?

Are the General Requirements in Sections 1.3 to 1.7 an adequate and consistent high level summary of the detailed requirements in Sections 3-7?

Definitions

Are the definitions in Section 2 complete?

Several of the terms in this section are defined differently in various sources. Where there was a choice, the project team gave priority to the definition currently used by ASME in risk informed Code Cases. Are any of the proposed definitions technically incorrect? If there are significant differences from definitions used in other sources, is there a quantifiable negative impact that would result from using the definition in the draft standard?

Risk Assessment Technical Requirements

Recognizing that the PRA described in this standard is meant to provide a reference for comparison with a plant PRA to be used in a variety of risk informed applications, does the PRA in this section provide a "realistic estimation of core damage frequency?" If not, what changes should be made? It would be most helpful if reviewers could provide specific wording where possible.

The project team would like feedback on the use of the words "shall," should," and "may." In an ASME document, the word "shall" indicates that a statement is a requirement to be understood as mandatory and leaves no decision to be made by the reader. The word "should" indicates that a statement is a recommendation, the advisability of which depends on the facts in each situation. The word "may" means that a recommendation is to be taken entirely at the reader's option. If a reviewer would like to suggest a change to the current draft standard, for example, that a requirement (use of "shall") be made a recommendation (use of "should" or "may"), the reviewer should provide sufficient explanation to help the project team evaluate the options.

There are places in the current draft where the project team would like feedback on a proposed parameter value. For example, Section 3.3.5.3.5 provides for an exception to the use of plant specific equipment failure rate data based on two numeric criteria. Again, where a reviewer would like to propose alternates to suggested parameter values, a detailed rationale and use of examples would be most helpful.

Documentation, configuration control, and peer review

Again, the project team would like feedback on the use of the words "shall," "should" and "may" in Sections 3-6 of the draft standard.

Where there are already established methods in these areas, for example the PSA certification peer review process in use by NSSS Owners Groups, the project team would like feedback on the compatibility of the requirements in this draft of the standard with those existing methods. For example, it would be helpful to have quantitative estimates of the impacts of any significant differences, along with suggestions for wording changes to the draft that would accomplish the same purpose but minimize any unnecessary negative impacts on potential users of the standard.

The risk assessment application process

Is the process described in Section 7 clear and logical?

What would be the impact on a user of this standard in applying this process to a variety of risk informed applications?

For a relatively simple risk informed application (say, risk ranking of components), how easy or cumbersome would this process be? For example, how many determinations of "sufficiency" and "significance" would have to be made, and what is the resource impact on a potential user?

How practical and resource-intensive would this process be in a more complicated risk informed application?

Does the process as currently described in this draft of the standard provide a reasonable amount of repeatability and consistency among potential users and regulatory reviewers?

It would be helpful to have feedback based on trial use of this process in actual risk informed applications.

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PREDECISIONAL DRAFT

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DRAFT

STANDARD FOR PROBABILISTIC RISK --ASSESSMENT FOR NUCLEAR POWER PLANT APPLICATIONS

Rev 14A May11, 2001

Date of Issuance: xxxxxxxx, xx, 2001

The 2001 edition of this Standard is being issued with an automatic addenda subscription service. The use of addenda allows revisions made in response to public reviewcomments or committee actions to be published on a regular yearly basis; revisions published in addenda will become effective six months after the Date of Issuance of the addenda. The next edition of this Standard is scheduled for publication in 2004.

ASME issues written replies to inquiries concerning interpretations of technical aspects of this Standard. The interpretations will be included with the above addenda service. Interpretations are not part of the addenda to the Standard.

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FOREWORD

(This Foreword is not part of ASME PRA-S-2001)

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 in regards 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 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) was comprised of 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. 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 and all reviews, the draft Standard was submitted to the consensus technical standards committee, the Committee on Nuclear Risk Management (CNRM), for approval 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.

CNRM operates under procedures accredited by the American National Standards Institute as meeting the criteria of consensus procedures for American National Standards. It was approved by the ASME Board on Nuclear Codes and

Standards and subsequently approved by the American National Standards Institute on XXXXXX XX, 2001.

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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 requirements, or to the consideration of revisions to the present 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, propriety 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 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.

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COMMITTEE ON NUCLEAR RISK MANAGEMENT

(As of June 11, 2001)

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American Nuclear Society

CONTENTS

(A Detailed Contents Precedes Each Section)

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SECTIONS

SUMMARY OF CHANGES

The 2001 edition of this Standard is the first publication approved by the Committee on Nuclear Risk Management and ASME, and after public review, ASME PRA-S–2001 was approved by the American National Standards Institute on XXXXX XX, 2001.

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PREFACE

(This Preface is Not Part of ASME PRA-S-2001)

ORGANIZATION OF THIS STANDARD

This Standard is organized into Sections as follows:

Sections

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

Each Section is subdivided into Subsections, Paragraphs, and with Bullets and Sub-bullets identified as follows:

Examples

Subsections	=	3.1
Paragraphs	=	3.1.2

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

Bullets	=	•
Sub-bullets	=	⇒

Tables and Figures provided in this Standard are identified by the applicable Subsection, or Paragraph 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 located immediately following the Subsection, or Paragraph text that applies to its use.

References are identified sequentially within the text of each Paragraph as follows: [3.1.2-l], [3.1.2-2], or [3.1.2-3], etc., and then listed at the end of the Paragraph.

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

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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 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 describes 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.

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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.

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 100 16-5990.

ADDENDA SERVICE

This edition of ASME PRA-S-2001 includes an automatic addenda subscription service 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.

1 INTRODUCTION

Contents

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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 LERF 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 Subsection 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 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. While 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-l describes the attributes of a PRA which were used to develop Section 4.0 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 **Level-2/LERF PRA.** They and their abbreviations are as follows:

Initiating Events Analysis	**(IE)**
Accident Sequence Analysis	(AS)
Success Criteria	**(SC)**
Systems Analysis	**(SY)**
Human Reliability Analysis	**(HR)**
Data Analysis	**(DA)**
Internal Flooding	**(IF)**
Quantification	**(QU)**
LERF Analysis	**(LE)**

1.4.2 High Level Requirements. A set of Objectives and **HLRs** are provided for each PRA Element in tables of Section. 4. All **PRAs** using this Standard shall satisfy each of **these HLRs,** but to differing degrees, as explained in Subsection 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 tables of Section 4 as action statements, using the three Capability Categories described in Subsection 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-l 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.

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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 from application to application. 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 Subsection 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.

CRITERIA	CAPABILITY CATEGORY I	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III
1. Scone and level of detail: The degree to which resolution and specificity are incorporated such that the technical issues are addressed	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 contributors at the SSC level including associated human actions, as necessary [see Note (1)]	Resolution and specificity sufficient to identify the relative importance of the contributors at the component level including associated human actions, as necessary [see Note (1)]
2. Plant-snecificitv: 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 to capture to the extent practical all significant features represented in the scope of the PRA model	Use of plant-specific data/models to capture to the extent practical all significant features represented in the scope of the PRA model
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 [see Note (2)]	Departures from realism will have small impact on the conclusions and risk insights as supported by good practices [see Note (2)]	Departures from realism will have negligible impact on the conclusions and risk insights as supported by good practices [see Note (2)]

NOTES:

(1) The definition for Capability Category II is not meant to imply that the resolution and specificity is to a level to identify every SSC and human action; only those necessary for the specific SR. Similarly for Capability Category III, it is not meant to imply that the resolution and specificity is to a level to identify every sub-component for every component.

(2) Differentiation from moderate (conservative or acknowledged, potential non-conservative), to small, to negligible is determined by the extent to which the impact on the conclusions and risk insights could affect a decision under consideration. This differentiation recognizes that the PRA would generally not be the sole input to a decision. A moderate impact implies that the impact (of the departure from realism) is of sufficient size that it is likely that a decision could be affected; a small impact implies that it is unlikely that a decision could be affected, and a negligible impact implies that a decision would not be affected.

Table 1.3-i **BASES FOR PRA CAPABILITY CATEGORIES**

2 DEFINITIONS

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

ACRONYMS

AOT - Allowed Outage Time
ADS - Automatic Depressurization System
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
RCP - Reactor Coolant Pump
RCIC - Reactor Core Isolation Cooling
RCS - Reactor Coolant System
RPT - Reactor Pump Trip
RPV - Reactor Pressure Vessel
RWST - Refueling Water Storage Tank
SAR - Safety Analysis Report
SLCS - Standby Liquid Control System
SBO - Station Blackout
SGTR - Steam Generator Tube Rupture
SORV - Stuck Open Relief Valve
SSCs - Structures, Systems, and Components
SR - Supporting Requirements
SW - Service Water
THERP - Technique For Human Error Rate Prediction (see NUREG/CR-1278)
TS - Technical Specifications

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 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 – availability is 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

diagnosis – examination and evaluation of data to determine either the condition of a 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 quantifiable, logical network 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)

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

failure mode - a condition or degradation mechanism that precludes the successful operation of a piece of equipment, a component, or a system

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 of time expressed as the ratio of the number of failures to a selected unit of time

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 - *an* engineered safety system used to provide core or containment cooling, reactivity control or pressure control, and to prevent cork damage, reactor coolant system failure, or containment failure

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 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, 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

human failure event (HFE) – an integrated logic description of HEPs based on the error model, performance shaping factor assessment, and other qualitative information needed to justify a single input to the risk model

human reliability analysis (HRA) – a structured approach used to identify potential human failure events and to systematically estimate the probability of those errors 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, operator errors, or both, 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

key safety functions – are 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 bypass 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

Level 1 analysis - identification and quantification of the sequences of events leading to the onset of core damage

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

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 me 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 or human error in order to mitigate or reduce the consequences of the failure

recovery models - types of Human Reliability Models that represent the act, process, or instance of recovering as a probability for use in a fault tree, event tree or cutset

required time - the time that is 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 - to react 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.

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

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

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 parameter used -by some PRA analysis techniques when quantifying an event tree. It represents the . relative frequency or degree-of-belief that each possible outcome, or branch, of a particular top event may be expected to occur

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*** quantification 'of the imprecision of the PRA which identifies the sources of uncertainty in the PRA model and characterizes their impact on the overall results of a PRA ***(i.e., CDF or LERF)***

walkdown – inspection of local areas in a nuclear power plant where systems and components are physically located in order 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.

3. RISK ASSESSMENT APPLICATION PROCESS

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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. 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. 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. 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. 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. 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 Figure 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 High Level Requirements and SRs for a specified set of Capability Categories. The determination of how the PRA is used in the application and

which Capability Categories are appropriate for each application must be made on a case-by-case basis.

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

3.2.1 Identification of Application. Define the application by:

- evaluating the plant design or operational change being assessed (Box 1 of Figure 3.1-1).
- identifying the SSCs and plant activities affected by the change including the cause-effect relationship between the plant design or operational change and the PRA model (Box 2 of Figure 3.1-1).
- identifying the PRA scope and PRA risk metrics that are needed to assess the change (Box 3 of Figure 3.1-1).

References [3.2-1] and [3.2-2] 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.*

In order 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 a 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.

References

[3.2-1] True, D., et. al; PSA Applications Guide, EPRI Report TR-105396, August 1995.

[3.2-2] Use of Probabilistic Risk Assessment in Plant-Specific, **Risk-informed** Decision-making: General Guidance, NRC Standard Review Plan, Chapter 19, NUREG-0800, 1998.

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 Subsection 1.3. For the application, determine the Capability Category for each part of the PRA needed to support the

application (Box 4 of Figure 3. 1-l). 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; and

(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 **$2x10^{-5}$/yr** and **$1x10^{-6}$/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 IL whereas the remaining parts of the PRA needed to determine CDF are determined to only require PRA Capability 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 Figure 3. 1-l). 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 Figure 3.1 **-1),** or generate supplementary analyses (See Subsection 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: (I) 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 Figure 3. l-l). 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 Figure 3.1 **-1),** or generate supplementary analyses (See Subsection *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 which 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, 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% 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 Paragraph 3.2.2, are sufficient to assess the application under consideration (Box 8 of Figure **3.1-1).**

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

Example: For the example discussed in Subsection 3.3, the scope of PRA elements defined in Section 4 of this Standard are sufficient and adequate to assess the plant change.

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 Figure 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 1.1 of Figure 3. 1-l). The bases for this determination shall be documented.

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

(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 Figure 3.1- 1), or generate supplementary analyses (See Subsection 3.6). Any upgrade of the PRA shall be done and documented in accordance with Section 5.

Example: The examples provided under Subsection 3.3 are applicable.

3.6 Use of Supplementary Analyses/Requirements (Stage E)

In the event that the scope of either the PRA or the standard is insufficient, supplementary analyses or requirements may be used (Box 7 of Figure 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 a LERF condition. This example illustrates a process of addressing SSC model adequacy by using general risk information 'to support the placement of MOVs into the appropriate risk category.

Supplementary requirements shall be drawn from other recognized codes or standards whose scopes complement that of this standard and which 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 ofpublished 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 o f 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 reference 3.5-1, evaluation offailure 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 that 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 reference 3.6-2, may be used. By using the requirements in reference 3.6-2, the seismic capacity of the commercial-grade valves can be evaluated and can be compared to that of thk 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 Figure 3. l-l). 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 Figure 3.1-1). Such supplementary analyses/requirements are outside the scope of this Standard.

References

[3.6- 1] "Requirements for Safety Significance Categorization of Snubbers using Risk Insights and Testing Strategies for Inservice Testing of LWR Power Plants," ASME Code for Operation and Maintenance of Nuclear Power Plants-Code Case OMN-10.

[3.6-2] Draft "External Events PRA Methodology Standard," American Nuclear Society Standard, ANS-58.21.

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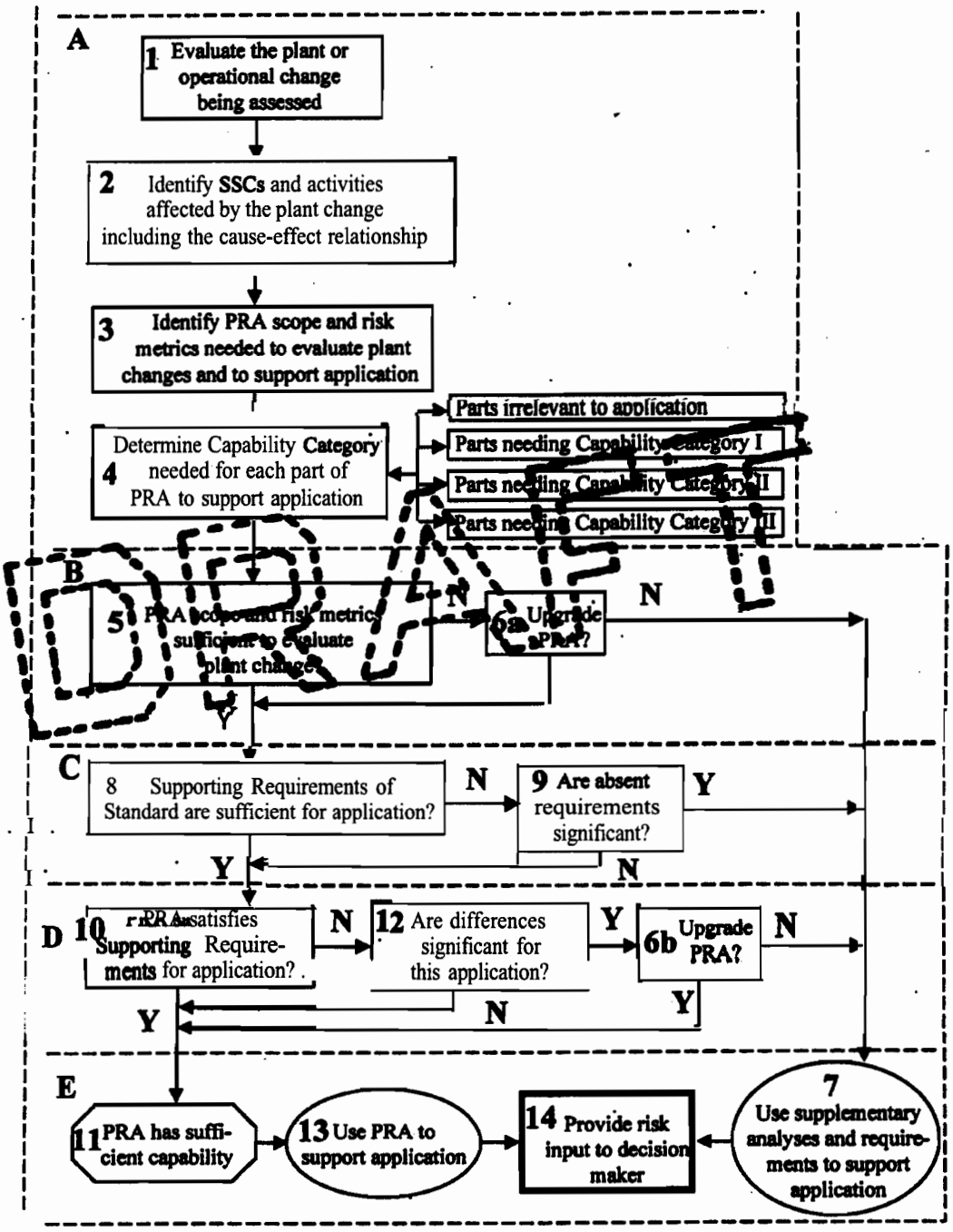

FIG. 3.1- 1 Application Process Flowchart

4 RISK ASSESSMENT TECHNICAL REQUIREMENTS

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Objectives and Tables

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 in 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., PRA data analysis) or used to support the PRA (e.g., thermal-hydraulic 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 Subsection provides requirements for the use of expert judgment outside of the PRA analysis team to resolve a specific technical issue.

NUREG/CR-6372 (Reference **[4.3-1]**) and NUREG-1563 (Reference **[4.3-2]**) may be used to meet the requirements in this Subsection. 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 (Reference [434-3]) and the Yucca Mountain project's study of volcanic hazards (Reference [4.3-4]). 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 (Reference [4.3-1]).

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

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 or experts.

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:

- Complex experimental data exist that the analysts **know** have been interpreted differently by different outside experts
- More than one conceptual model exists for interpreting the technical issue, and judgment is needed as to the applicability of the different models

- Judgments are required to assess whether bounding assumptions or calculations are appropriately conservative
- 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; and

(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; and

(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,

References

[4.3-1] R.J. Budnitz, G. Apostolakis, D.M. Boore, L.S. Cluff, K.J. Coppersmith, C.A. Cornell, and P.A. Morris, "Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on the Use of Experts", U.S. Nuclear Regulatory Commission and Lawrence Livermore National Laboratory, Report NUREG/CR-6372, 1997

[4.3-2]J.P. Kotra, M.P. Lee, N.A. Eisenberg, and A.R. DeWispelare, "Branch Technical Position on the Use of Expert Elicitation in the High-Level, Radioactive Waste Program", U.S. Nuclear Regulatory Commission, Office of Nuclear Materials Safety and Safeguards, Report NUREG-1563, 1996

[4.3-3] Pacific Gas and Electric Company,, "Final Report of the Diablo Canyon Long-Term Seismic Program", US Nuclear Regulatory Commission Docket numbers 50-275 and 50-323.

[4.3-4] Geomatrix, "Probabilistic Volcanic Hazards Analysis for Yucca Mountain, Nevada", U..S. Department of Energy Yucca Mountain Project, Report BA000-1717-2200-00082, 1996

4.4 PRA Requirements

4.4.1 Derivation. Objectives 'were established for each of the nine elements used to characterize a PRAThe Objectives reflect substantial experience accumulated with PRA development and usage, and are consistent with the PRA Procedures Guide (Reference [4.4. l-l]) and the NEI-00-02 Peer Review Process Guidance (Reference [4.4.1-

23). 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 High Level Requirements 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 Supporting Requirements. 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 High Level Requirements 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 High Level Requirements generally address attributes of the PRA Element such as:

- **Scope and level of detail**
- model fidelity and realism
- output or quantitative results (if applicable)
- 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 Subsection 1.3

4.4.2, Requirements. Tables 4.4-l through 4.4-9 list the **HLRs** and **SRs** for each of the nine PRA Elements. Each Table is preceded by a statement of the Objectives for the Elements. 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-l '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.

References

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

[4.4.1-2] Probabilistic Risk Assessment Peer Review Process Guidance, NEI-00-02, March 2000

4.4.1 INITIATING EVENT ANALYSIS

OBJECTIVE& The objective of the initiating event analysis is : to identify and quantify events that could lead to core damage in such a way that:

- Events that challenge normal plant operation and that require successful mitigation to prevent core damage are included.
- Initiating events are grouped according to the mitigation requirements to facilitate the effkient modeling of plant response.
- Frequencies of the initiating event groups are quantified.

TABLE 4.4-1 HIGH LEVEL REQUIREMENTS FOR INITIATING EVENTS ANALYSIS (HLR-IE)

A The initiating event analysis shall provide a reasonably complete identification of initiating events.	
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.	
C The initiating event analysis shall estimate the annual frequency of each initiating event or initiating event group.	
D The initiating event analysis shall be documented in a manner that facilitates review by describing the processes that were followed to select, group, and model and quantify the initiating event frequencies, with assumptions and bases stated.	PRA applications, upgrades, and peer screen the initiating event list and to

TABLE 4.4-1a SUPPORTING REQUIREMENTS FOR INITIATING EVENTS ANALYSIS HIGH LEVEL REQUIREMENT A, IDENTIFICATION

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.*		

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TABLE 4.4-1a SUPPORTING REQUIREMENTS FOR INITIATING EVENTS ANALYSIS HIGH LEVEL REQUIREMENT A, IDENTIFICATION.

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: Transients INCLUDE among the transients both equipment and human induced events that disrupt the plant and leave the primary system pressure boundary intact. 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: Small LOCAs *Examples: reactor coolant pump seal LOCAs, small pipe breaks* Medium LOCAs *Examples: stuck open safety or relief valves* Large LOCAs *Examples: inadvertent ADS, component ruptures* Excessive LOCAs (LOCAs that cannot be mitigated by any combination of engineered systems) *Example: reactor pressure vessel rupture* LOCAs Outside Containment *Example: pipe breaks outside containment* 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)]. Special initiators * (e.g., support systems failures, instrument line breaks) Internal flooding initiators (see IF-D1 and D2)* * These initiators may result in either a transient or a LOCA type of sequence		

TABLE 4.4-1a SUPPORTING REQUIREMENTS FOR INITIATING EVENTS ANALYSIS HIGH LEVEL REQUIREMENT A, IDENTIFICATION

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-A3	REVIEW the plant-specific initiating event experience of all initiators to assure 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.		
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 qualitative review of system impacts to identify potentially risk-significant system initiating events. INCLUDE initiating events resulting from multiple failures, if the equipment failures result from a common cause.	PERFORM a systematic evaluation of each system to assess the possibility of an initiating event occurring due to a failure of the system. USE a structured approach (such as a system-by-system review of initiating event potential, or an FMEA [failure modes and effects analysis] or fault tree) to assess and document the possibility of an initiating event resulting from system failures. INCLUDE initiating events resulting from multiple failures, if the equipment failures result from a common cause..	PERFORM a systematic evaluation of each system to assess the possibility of an initiating event occurring due to a failure of the system. 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.
IE-AS	In the identification of the initiating events, INCORPORATE (i) events that have occurred at conditions other than at-power operation (i.e., during low-power or shutdown conditions); and (ii) 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.		

TABLE 4.4-1a SUPPORTING REQUIREMENTS FOR INITIATING EVENTS ANALYSIS HIGH LEVEL REQUIREMENT A, IDENTIFICATION

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-A6		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			In searching for initiating events, ACCOUNT FOR initiating event precursors, both to help identify initiating events, and to provide a partial basis for quantifying their frequencies.
IE-A8		In searching for initiating events, ACCOUNT FOR each system alignment and alignments of supporting systems that could influence the likelihood that failures cause an initiating event, or magnify the severity of the challenge to plant safety functions that would result from such an event.	
IE-A9		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.		

TABLE 4.4-1b SUPPORTING REQUIREMENTS FOR INITIATING EVENTS ANALYSIS HIGH LEVEL REQUIREMENT B, GROUPING

The initiating event'anaiysis 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 (Section 4.4.2) and to facilitate quantification in the *Quantification* element (Section 4.4.8). 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).*		

TABLE 4.4-1b SUPPORTING REQUIREMENTS FOR INITIATING EVENTS ANALYSIS HIGH LEVEL REQUIREMENT B, GROUPING

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 and operator response, success criteria, timing, 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.	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.

TABLE 4.4-1b SUPPORTING REQUIREMENTS FOR INITIATING EVENTS ANALYSIS HIGH LEVEL REQUIREMENT B. GROUPING

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-B4	GROUP separately from other initiating event categories those categories with significantly different plant response impacts or which 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 4.4-1c SUPPORTING REQUIREMENTS FOR INITIATING EVENTS ANALYSIS HIGH LEVEL REQUIREMENT C. QUANTIFICATION

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 I I	CAPABILITY CATEGORY III
IE-Cl	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.		
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-Cl), 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 which could be different from historical values.		

TABLE **4.4-1c** SUPPORTING REQUIREMENTS FOR INITIATING EVENTS ANALYSIS HIGH **LEVEL REQUIREMENT C. QUANTIFICATION**

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-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 (Subsection 4.4.6) and the *Level-1-quantification* section (Subsection 4.4.8).		
IE-C5			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 Reference [4.4.1-1].
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 (Subsection 4.4.4).		

TABLE 4.4-1c SUPPORTING REQUIREMENTS FOR INITIATING EVENTS ANALYSIS HIGH LEVEL REOUIREMENT C, QUANTIFICATION

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, Subsection 4.4.4.). 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, Section 4.4.6, 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		If fault-tree modeling is used, USE plant-specific information in the assessment and quantification of recovery actions where available. See *Human Reliability Analysis* (Subsection 4.4.5) 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.		

TABLE 4.4-1c SUPPORTING REQUIREMENTS FOR INITIATING EVENTS ANALYSIS HIGH LEVEL REOUIREMENT C. OUANTIFICATION

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 Section *4.3, Use of Expert Judgment, as* appropriate. For purposes of this Requirement, a "rare event" is au event that might be expected to occur one or a few times throughout the world nuclear industry over many years. An "extremely rare event" is an event that would not be expected to occur even once throughout the industry over many years.		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 Section *4.3, Use of Expert Judgment,* as appropriate. For purposes of this Requirement, a "rare event" is au event that might be expected to occur one or a few times throughout the world nuclear industry over many years. An "extremely rare event" is an event that 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:* *Plant geography, climate, and meteorology for LOOP and LOOP recovery* *Service water intake characteristics and plant experience* *LOCA frequency calculation*

TABLE 4.4-1c SUPPORTING REQUIREMENTS FOR INITIATING EVENTS ANALYSIS HIGH LEVEL REOUIREMENT C, QUANTIFICATION

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 I I	CAPABILITY CATEGORY III
IE-C12	In the interfacing system LOCA frequency analysis, INCLUDE those features of plant and procedures that could significantly influence the ISLOCA frequency.		In the ISLOCA frequency analysis, INCLUDE those features of plant and procedures that could significantly influence the ISLOCA frequency: · EVALUATE surveillance procedure steps · INCLUDE surveillance test intervals explicitly · ASSESS on-line surveillance testing quantitatively · QUANTIFY pipe rupture probability · ADDRESS explicitly valve design (e.g., air operated testable check valves) · INCLUDE quantitatively the valve isolation capability given the high-to-low-pressure differential.

Reference

[4.4-1-l]: NUREG/CR-5750, "Rates of Initiating Events at U.S. Nuclear Power Plants", Idaho National Engineering and Environmental Laboratory, Idaho Falls, February 1999

TABLE 4.4-1d SUPPORTING REQUIREMENTS FOR INITIATING EVENTS ANALYSIS HIGH-LEVEL REQUIREMENT D. DOCUMENTATION

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 LIST and JUSTIFY functional categories considered, in accordance with the safety functions considered in the accident sequence model. For each functional category, DOCUMENT the specific initiating events considered. DOCUMENT the systematic search for plant unique and plant specific support system initiators, along with the resulting support system initiators disposition. DOCUMENT the systematic search for RCS pressure boundary failures and interfacing system LOCAs. DOCUMENT the approach for assessing completeness and consistency of initiating events with plant specific experience, industry experience, other comparable PRAs and FSAR initiating events. DOCUMENT the assumptions.		
IE-D2	Grouping and Screening of Initiating Events DOCUMENT the basis for screening out initiators as risk insignificant. DOCUMENT the basis for grouping and subsuming initiating events. This may interface with the required success criteria from the *Systems Analysis* section (Subsection *4.4.4)* and Success *Criteria* section (Subsection 4.4.3) of this Standard. DOCUMENT the assumptions. DOCUMENT the dismissal of any observed initiating events, including any credit for recovery.		

TABLE 4.4-1d SUPPORTING REQUIREMENTS FOR INITIATING EVENTS ANALYSIS HIGH-LEVEL REQUIREMENT D, DOCUMENTATION

The initiating event anal&is 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 th@initiating event list rind 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-D3	Quantification of Initiating Event Frequencies DOCUMENT the derivation of the initiating event: frequencies and the recoveries used in conjunction with the initiating event. DOCUMENT the approach to quantification of each initiating event frequency as data analysis or model approach. 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. 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. DOCUMENT the methodology and approach when using data analysis methods to estimate initiating event frequencies; also, IDENTIFY the data used. DOCUMENT the justification for exclusion of any data. DOCUMENT the basis for the availability factor used to convert initiating event frequencies to events per reactor year. IDENTIFY potential time dependent aspects of the initiating event frequencies, and DOCUMENT assumptions made to obtain average frequencies. DOCUMENT the process for computing initiating event frequencies. 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.		

4.4.2 ACCIDENT SEQUENCE ANALYSIS

OBJECTIVE: The objective of the accident sequence element is 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:

- 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.
- Plant-specific dependencies are reflected in the accident sequence structure.
- 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.
- End states are clearly defined to be core damage or successful mitigation with capability to support the Level 1 to Level 2 interface.

Table 4.4.2 HIGH LEVEL REQUIREMENTS FOR ACCIDENT SEQUENCE ANALYSIS (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.
B.	Dependencies due to initiating events, human interface, functional dependencies, environmental and spatial impacts, and common cause failures shall be addressed.
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 4.4-2a .

ACCIDENT SEQUENCE ANALYSIS SUPPORTING REQUIREMENTS (SR-AS):

HIGH-LEVEL REQUIREMENT 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 necessary 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 needed 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.		

TABLE 4.4-2a

ACCIDENT SEQUENCE ANALYSIS SUPPORTING REQUIREMENTS (SR-AS): HIGH-LEVEL REQUIREMENT 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-A9	JSE 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) hat 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 the critical safety function status 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.

TABLE 4.4-2a

ACCIDENT SEQUENCE ANALYSIS SUPPORTING REQUIREMENTS (SR-AS):

HIGH-LEVEL REQUIREMENT 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-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.		

TABLE 4.4-2b

ACCIDENT SEQUENCE ANALYSIS SUPPORTING REQUIREMENTS (SR-AS):

HIGH-LEVEL REQUIREMENT 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: • Turbine driven system dependency on SORV, depressurization and containment heat removal (suppression pool cooling). • Low pressure system injection success dependent on need for RPV depressurization.		
AS-B3	For each accident sequence, IDENTIFY the phenomenological conditions created by the accident progression. Phenomenological impacts include generation of harsh environments affecting temperature, pressure, debris, water levels, humidity, etc. that could impact the success of the system or function under consideration; for example, 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.		

TABLE 4.4-2b
ACCIDENT SEQUENCE ANALYSIS SUPPORTING REQUIREMENTS (SR-AS):
HIGH-LEVEL REQUIREMENT 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-B6	INCLUDE events for which time phased dependencies might exist. For SBO/LOOP sequences, INCLUDE key time phased events such as: • AC power recovery • DC battery adequacy (time dependent discharge) • Environmental conditions (e.g., room cooling) for operating equipment and the control room For ATWS/failure to scram events (for BWRs). INCLUDE key time dependent actions such as: • SLCS initiation • RPV level control • ADS inhibit Other events that may be subject to explicit time dependent characterization include: • CRD as an adequate RPV injection source Long term make-up to RWST		

TABLE 4.4-c --
ACCIDENT SEQUENCE ANALYSIS SUPPORTING REQUIREMENTS (SR-AS):
HIGH-LEVEL REQUIREMENT 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 [AS-25]	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 [AS-26]	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 link between the definition of initiating event category *in the Initiating Event Analysis* section and the event sequence model. • The definition of core damage and associated success criteria that is consistent with that documented in the Success Criteria Task. • Key definitions of operator actions and sequence specific timing and dependencies reflected in the event trees that is traceable to the HRA for these actions. • A description of the interface of the accident sequence models with plant damage states • A framework for an integrated treatment of dependencies in the Initiating Events Analysis, Systems Analysis, Data Analysis, Human Reliability Analysis and Level 1 Quantification.		

TABLE 4.4-2c

ACCIDENT SEQUENCE ANALYSIS SUPPORTING REQUIREMENTS (SR-AS):

HIGH-LEVEL REQUIREMENT 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-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 co **mponents 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, as well as 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 "no **rmal** design parameters and for which credit has been taken; and *(h)* how all requirements for *Accident Sequence Analysis have* been **sati sfied when sequenc**es are modeled using a single top event linked fault tree.		

4.4.3 SUCCESS CRITERIA

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

- Overall success criteria are defined (i.e., core damage and large early release).
- Success criteria are defined for critical safety functions, supporting systems, structures, components and operator actions necessary to support accident sequence development.
- The methods and approaches have a firm technical basis.
- The resulting success criteria are referenced to the specific deterministic calculations.

Table 4.4-3 HIGH LEVEL REQUIREMENTS FOR SUCCESS CRITERIA AND SUPPORTING ENGINEERING CALCULATIONS (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.
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.
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 4.4-3a .

SUPPORTING REQUIREMENTS FOR SUCCESS CRITERIA AND OTHER ENGINEERING CALCULATIONS HIGH LEVEL REQUIREMENT 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: · IDENTIFY any substantial differences from the Section 2 definition, and · 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 liniits and code-calculated values to allow for limitations of the codes, sophistication of th models, and uncertainties in the results. *Examples of measures for core damage that have been used in PRAs include:* • *Collapsed liquid level less than 1/3 core height OR code-predicted peak core temperature > 2500°F (BWR)* • *Collapsed liquid level below top of active fuel for a pro longed period, OR code-predicted core peak node temperature > 2200°F using a code with detailed core modeling, OR code-predicted core peak node temperature > 1800°F using a code with simplified (e.g., single-node core model, lumped parameter) core modeling, OR code-predicted core exit temperature > 1200°F for 30 minutes using a code with simplified core modeling (P WR)*		
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.		

Table 4.4-3a

SUPPORTING REQUIREMENTS FOR SUCCESS CRITERIA AND OTHER ENGINEERING CALCULATIONS HIGH LEVEL REQUIREMENT 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-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.		
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 hours. Mission times for individual SSCs that function during the accident sequence may be less than 24 hours, 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 hours 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:* • *assign an appropriate plant damage state for the sequence;* • *extend the mission time, and adjust the affected analyses, to the point at which conditions can be shown to reach acceptable values; or* • *model 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 success@ 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.		

Table 4.4-3b .

SUPPORTING REQUIREMENTS FOR SUCCESS CRITERIA AND OTHER ENGINEERING CALCULATIONS HIGH-LEVEL REQUIREMENT 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, deter&nation 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 an appropriate combination of plant-specific or generic, **best-**estimate or conservative analyses/evaluations that **are** applicable to the plant.	**USE an appropriate combination of best-estimate plant-specific or 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	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. Best-estimate plant-specific models or analyses may be supplemented with FSAR or generic analysis, but ***only if* such** supplemental analyses are applicable to the plant and do not affect risk significant **CDF/LERF** sequences. .

Table 4.4-3b

SUPPORTING REQUIREMENTS FOR SUCCESS CRITERIA AND OTHER ENGINEERING CALCULATIONS HIGH LEVEL REQUIREMENT 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-B2	MINIMIZE the use of expert judgment in situations where applicable analysis results exist, or situations where analysis tools exist and can reasonably be employed.		DO NOT USE expert judgment in situations where applicable analysis results exist, or situations where analysis tools exist and can reasonably be employed.
SC-B3	When defining success criteria, USE thermal/hydraulic, structural, or other analyses/evaluations appropriate to the event being analyzed. *Examples include:* • *engineering calculations;* • *computer codes with detailedplant models;* • *results of tests with conditions corresponding to the accident sequences;* • *results of generic or plant-specific analyses for similar transients where these are shown to be appropriate.*		When defining success criteria, USE scenario-specific thermal./ hydraulic, structural, or other analyses/evaluations appropriate to the event being analyzed. *Examples include:* • *engineering calculations;* • *computer codes with detailed plant models;* • *results of tests with conditions corresponding to the accident sequences;.* • *results of plant-specific analyses for similar transients where these are shown to be appropriate.*

Table 4.4-3b

SUPPORTING REQUIREMENTS FOR SUCCESS CRITERIA AND OTHER ENGINEERING CALCULATIONS HIGH LEVEL REQUIREMENTB

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-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:* · ***COMPARE with results of the same analyses performed for similar plants, accounting for differences in unique plant features;*** · ***COMPARE with results of similar analyses performed with other plant-specific codes;*** · ***CHECK by other means appropriate to the particular analysis.***		
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 4.4-3c

SUPPORTING REQUIREMENTS FOR SUCCESS CRITERIA AND OTHER ENGINEERING CALCULATIONS

HIGH LEVEL REQUIREMENT 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. . 'IDENTIFY conservative, optimistic, or simplifying assumptions or conditions • 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. • PROVIDE the basis for the success criteria development process and the supporting engineering calculations.	
SC-C2	DOCUMENT uses of expert judgment	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.		

Table 4.4-3c

SUPPORTING REQUIREMENTS FOR SUCCESS CRITERIA AND OTHER ENGINEERING CALCULATIONS

HIGH LEVEL REQUIREMENT C

Documentation shrill 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 deta ils of the assumptions made and their bases. (HLR-SC-C)

Index No. SC-C	CAPABILITY CATEGORY I	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III
SC-C4	DOCUMENT the following, to the extent they are not included in the documentation under AS-C4 and SY-C1: • 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 rea ctor vessel level)*; • calculations (generic and plant-specific) or other references used to establish success criteria, and identification of cases for which they are used; • identification of computer codes or other methods used to establish plant-specific success criteria; • 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; • identification of important assumptions used in establishing success criteria; • a summary of success criteria for the available mitigating systems and human actions for each accident initiating group modeled in the PRA; • the basis for establishing the time available for human actions; • descriptions of processes used to define success criteria for grouped initiating events or accident sequences.		

4.4.4 SYSTEMS ANALYSIS

OBJECTIVE: The objective of the systems analysis element is 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:

- 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.
- 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.
- Different initial system alignments are evaluated to the extent needed for CDF and LERF determination.
- 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 4.4-4 HIGH LEVEL REQUIREMENTS FOR SYSTEMS ANALYSIS (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.
B	The systems analysis shall provide a reasonably complete treatment of common cause failures and intersystem and intra-system dependencies.
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 **4.4-4a**

SUPPORTING REQUIREMENTS FOR SYSTEMS ANALYSIS HIGH LEVEL REQUIREMENT 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 sequence analyses.		
SY-A2	COLLECT pertinent information to ensure that the system 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.*		

TABLE 4.4-4a

SUPPORTING REQUIREMENTS FOR SYSTEMS ANALYSIS HIGH LEVEL REQUIREMENT 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-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) procedures for the operation of the system during normal and abnormal conditions; and (g) system configuration during normal and abnormal conditions.		
SY-A4		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.		

TABLE 4.4-4a

SUPPORTING REQUIREMENTS FOR **SYSTEMS ANALYSIS** HIGH LEVEL **REQUIREMENT 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-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 several failures are combined into super 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.	

TABLE 4.4-4a

SUPPORTING REQUIREMENTS FOR SYSTEMS ANALYSIS HIGH LEVEL REQUIREMENT 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-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:* • *hardware failures that are not recoverable versus actuation signals which are recoverable* • *HE events that can have different probabilities dependent on the context of different accident sequences* • *events which are mutually exclusive of other events not in the module* • *events which occur in other fault trees (especially common-cause events)* • *SSCs used by other systems*		
SY-A11	INCORPORATE the effect of variable success criteria into the system modeling. *Example causes of variable system success criteria are:* • *Different accident scenarios -- different success criteria are requiredfor 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* • *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)* • *Time dependence -- success criteriafor 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 requiredfor mitigation later following the accident).*		

TABLE 4.4-4a

SUPPORTING REQUIREMENTS FOR SYSTEMS. ANALYSIS HIGH LEVEL REOUIREMENT 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 HI
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-A15. This equipment includes both active components (e.g., pumps, valves, and air compressors) and passive components (e.g., piping, heat exchangers, and tanks) required for system operation. 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.*		

TABLE 4.4-4a

SUPPORTING REQUIREMENTS FOR SYSTEMS ANALYSIS HIGH LEVEL REQUIREMENT 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-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 (not a comprehensive list):* • *active component fails to start* • *active componentfails to continue to run* • *failure of a closed component to open* • *failure of a closed component to remain closed* • *failure of an open component to close* • *failure of an open component to remain, open* • *active component spurious operation* • *plugging of an active or passive component* • *leakage of an active or passive component* • *rupture of an active or passive component* • *internal leakage of a component* • *internal rupture of a component* • *failure to provide signal/operate (e.g., instrumentation)* • *spurious signal/operation* • *pre-initiator human failure events*		

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TABLE 4.4-4a

SUPPORTING REQUIREMENTS FOR SYSTEMS ANALYSIS HIGH LEVEL REQUIREMENT 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-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 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. • One or more failure modes for a component may be excluded from the system's 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, or • 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, or • 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.		

TABLE 4.4-4s

SUPPORTING REQUIREMENTS FOR **SYSTEMS ANALYSIS** HIGH LEVEL **REQUIREMENT 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-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.)*		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.*)
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* and *Accident Sequence Analysis.*)		
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:* • *system-relatedparameters such as a high temperature within the system;* • *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)* • *adverse environmental conditions.*		

TABLE **4.4-4a**

SUPPORTING REQUIREMENTS FOR SYSTEMS ANALYSIS HIGH LEVEL REQUIREMENT 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-A18	**In the systems model, INCLUDE out-of-service unavailability for components due to testing and maintenance, unless screened, consistent with the actual practices** and history of the plant for removing equipment from service.	**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** INCLUDE: • **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. • maintenance events at **the train level** when procedures require isolating the entire train for maintenance. • **maintenance events at a sub-train level (i.e., between tagout boundaries, such as a functional equipment group) when directed by procedures.** ***Examples of out-of-service unavailability to be modeled:*** • ***Train outages during a work window for preventive/corrective maintenance*** • ***A functional equipment group (FEG) removed from service for preventive/corrective maintenance*** • ***A relief valve taken out of service***	

TABLE 4.4-4a

SUPPORTING REQUIREMENTS FOR SYSTEMS ANALYSIS HIGH LEVEL REQUIREMENT 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-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: • test or operational data • calculations • vendor input • 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	In the support-state approach, ASSIGN support states to account properly for system dependencies on other systems.		

TABLE 4.4-4a

SUPPORTING REQUIREMENTS FOR SYSTEMS ANALYSIS HIGH LEVEL REQUIREMENT 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-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. *For example, see reference 4.4.4-2*		

References

[4.4.4-1] NUREG/CR-2728 Interim Reliability Evaluation Program Procedures Guide, March 3, 1983.
4.4.4-2 NSAC-161, Faulted Systems Recovery Experience, May 1992

TABLE 4.4-4b

SUPPORTING REQUIREMENTS FOR SYSTEMS ANALYSIS HIGH LEVEL REQUIREMENT 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. 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. An acceptable method is represented in NUREG/CR-5485 (Reference [4.4.4-2]). *Candidates for common-cause failures include, for example:* • *motor-operated valves* • *pumps* • *safety-relief valves* • *air-operated valves* • *solenoid-operated valves* • *check valves* • *diesel generators* • *batteries* • *inverters and battery chargers* • *circuit breakers*	

TABLE 4.4-4b

SUPPORTING REQUIREMENTS FOR SYSTEMS ANALYSIS HIGH LEVEL REQUIREMENT 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 HI
SY-B2	ESTABLISH common cause failure groups by using a logical, systematic process that considers similarity in: • service conditions • environment • design or manufacturer • maintenance. JUSTIFY the basis for selecting common cause component groups		
SY-B3	ACCOUNT explicitly for the modeled system's dependency on support systems in the modeling process. This may be accomplished by: • fault tree linking • dependency matrices that are translated into event tree structure, event tree logic rules, or conditional split fractions • an evaluation that demonstrates that excluding the dependency does not significantly affect the system model.		

Reference

[4.4.4-3] NUREG/CR-5485 Guidelines on Modeling Common-Cause Failures in Probabilistic Risk Assessment, November 20, 1998.

TABLE 4.4-4b .

SUPPORTING REQUIREMENTS FOR SYSTEMS ANALYSIS HIGH LEVEL REQUIREMENT 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-B4	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-B5	BASE support system modeling on the use of conservative success criteria and timing. Realistic criteria and timing may be used when available.	BASE support system modeling on realistic success criteria and timing, unless a conservative approach can be justified.	BASE support system modeling on realistic success criteria and timing.
SY-B6	IDENTIFY spatial and environmental hazards that may impact system operation and ACCOUNT for them in the system fault tree or the accident sequence evaluation. *For Example: Use results of plant walkdowns as a source of information and resolution of issues in the evaluation of their impacts.*		

TABLE 4.4-4b

SUPPORTING REQUIREMENTS FOR SYSTEMS ANALYSIS HIGH LEVEL REQUIREMENT 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-B8	INCLUDE explicit treatment of containment vent effects (BWRs) and containment failure effects on system operation in the consideration of possible hazards.		
SY-B9	When modeling a system, INCLUDE the support systems required for successful operation of the system for a required mission time. *Examples:* • *actuation logic,* • *support systems requiredfor control of components,* • *component motive power,* • *cooling of components* • *any other identified support function (e.g., heat tracing) necessary to meet the success criteria and associated systems.* Exceptions: The treatment of circular logic may require approaches that do not strictly comply with this criterion.		
SY-B10	INCLUDE support systems required to supply motive power for continuous and successful operation of components in accordance with the success criteria in the system model (e.g., AC power to a motor-driven pump).		

TABLE 4.4-4b

SUPPORTING REQUIREMENTS FOR SYSTEMS ANALYSIS HIGH LEVEL REQUIREMENT 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-B10	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-B11	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-B12	DO NOT USE proceduralized recovery actions as the sole basis for eliminating a support system from the model. However, INCLUDE these recovery actions in the model quantification.		
SY-B13	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).*		

TABLE 4.4-4b.

SUPPORTING REQUIREMENTS FOR SYSTEMS ANALYSIS HIGH LEVEL REQUIREMENT 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-B14	IDENTIFY SSCs that may be required to operate in conditions beyond their environmental qualifications. INCLUDE dependent failures of multiple SSCs that result from operation in these adverse conditions. *Examples of degraded environments include:* • *LOCA inside containment with failure of containment heat removal* • *safety relief valve operability (small LOCA, drywell spray, severe accident) (for BWRs)* • *steam line breaks outside containment* • *debris that could plug screens/filters (both internal and external to the plant),* • *heating of the water supply (e.g., BWR suppression pool, PWR containment sump) that could affect pump operability* • *loss of NPSH for pumps* • *steam binding of pumps.*		
SY-B15	INCLUDE operator interface dependencies across systems or trains, where applicable.		

TABLE 4.4-4c

SUPPORTING REQUIREMENTS FOR SYSTEMS ANALYSIS HIGH LEVEL REQUIREMENT C

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

IndexNo. SY-C	CAPABILITYCATEGORYI	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III

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TABLE 4.4-4c

SUPPORTING REQUIREMENTS FOR SYSTEMS ANALYSIS HIGH LEVEL REQUIREMENT C

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

SY-CI	**DOCUMENT the system model used in the PRA system t i ctions 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:* • *system function and operation under normal and emergency operations* • *system model boundary* • *system schematic illustrating all equipment and components necessary for system operation* • *information and calculations to support equipment operability considerations and assumptions* • *actual operational history indicating any past problems in the system operation* • *system success criteria and relationship to accident sequence models* • *human actions necessary for operation of system* • *reference to system-related test and maintenance procedures* • *system dependencies and shared component interface* • *component spatial information* • *assumptions or simplifications made in development of the system models* • *a list of all components andfailure modes included in the model, along with justificationfor any exclusion of components and failure modes* • *a description of the modularization process (if used)* • *records of resolution of logic loops developed during fault tree linking (if used)* • *results of the system model evaluations* • *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*

TABLE 4.4-4c

SUPPORTING REQUIREMENTS FOR SYSTEMS **ANALYSIS** HIGH LEVEL REQUIREMENT C

The systems **analysis** shall be documented in a manner that facilities PRA applications, upgrades, and peer view by. Describing the **process** 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-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.		

4.4.5 HUMAN RELIABILITY ANALYSIS[1]

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

- both pre-initiating event and post-initiating event activities, including those modeled in support system initiating event fault trees, are addressed,
- logic model elements are defined to represent the effect of such personnel actions on system availability/unavailability and on accident sequence development,
- plant-specific and scenario-specific factors are-accounted for , including those factors that; influence either what activities are of interest or human performance, and
- human performance issues are addressed in an integral way so that issues of dependency are captured.

Table 4.4-5 High Level Requirements

Pre-Initiator HRA

A - A systematic process shaii 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.

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.

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.

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

Table 4.4-5 High Level Requirements (cont'd)

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

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

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.

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 ERA

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.

(1) The following reference provides useful background information for Human Reliability Analyis:
D.T.Wakefield, G.W. Parry, G.W. Hannaman, A.J.Spurgin, "Sharp 1 -Revised Systematic Human Action Reliability Procedure" EPRI Report TP-101711 (1992)'

Table 4.4-5a 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.

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)).		

Table 4.4-5b 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.

Index No. HR-B	CAPABILITY CATEGORY I	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III
HR-B1	ESTABLISH rules for screening classes of activities from further consideration. *For 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.*	ESTABLISH rules for screening individual activities from further consideration. *For Example: Screen maintenance and test activities from further consideration only if:* • *equipment is automatically re-aligned on system demand, or* • *following maintenance activities, a post-maintenance functional test is performed that reveals misalignment, or* • *equipment position is indicated in the control room, status is routinely checked and realignment can be effected from the control room, or* • *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 4.4-5c 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.

index No. HR-C	CAPABILITY CATEGORY I	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III
EIR-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: • equipment to the desired standby or operational status • initiation signal or set point for equipment start-up or realignment • 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: • equipment to the desired standby or operational status • initiation signal or set point for equipment start-up or realignment • automatic realignment or power ADD failure modes discovered through the review 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 stand by systems.		

Table 4.4-5d. **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.

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[1] and ASEP[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		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), and (b) the quality of the human-machine interface, including both the equipment configuration, and instrumentation and control layout.	

Table 4.4-5d . 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.

Index No. HR-D	CAPABILITY CATEGORY I	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III
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, and** **(b)** USE the following information to assess the potential **for recovery of pre-initiator:** • post-maintenance or post-calibration tests required and **performed by procedure** • independent verification, using a written check-off list, **which verify component status following maintenance/testing** • original performer, using a written check-off list, makes a **separate check of component status at a later time** • 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.		

References:

(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)

Table 4.4-5e 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.

Index No. HR-E	CAPABILITY CATEGORY I	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III
HR-El	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, and (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 defmed 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);, and (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).	

Table 4.4-5e **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.

Index No. HR-E	CAPABILITY CATEGORY I	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III
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		USE simulator observations or talk-through with operators to confirm response models for dominant scenarios	USE simulator observations or talk-throughs with operators to confirm the response models for scenarios modeled.

Table 4.4-5f 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

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.		
HR-F2	COMPLETE THE DEFINITION of the HFEs by specifying: • accident sequence specific timing of cues, and time window for successful completion, and • accident sequence specific procedural guidance (e. g., AOPs, and EOPs), and • the availability of cues and other indications for detection and evaluation errors, and • the complexity of the response. (Task analysis is not required)	COMPLETE THE DEFINITION of the HFEs by specifying: • accident sequence specific timing of cues, and time window for successful completion, and • accident sequence specific procedural guidance (e. g., AOPs, and EOPs), and • the availability of cues and other indications for detection and evaluation errors, and • 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: • accident sequence specific timing of cues, and time window for successful completion, and • accident sequence specific procedural guidance (e. g., AOPs and EOPs), and • the availability of cues and other indications for detection and evaluation errors, and • the specific detailed tasks (e.g., component level) required to achieve the goal of the response.

Table 4.4-5g **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.**

Index No. HR-G	CAPABILITY CATEGORY I	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III
HR-G1	USE conservatwe estimates or PERFORM detailed analyses for the estimation of the HEPs of the HFEs in dominant accident sequences. Screening values may be used for the probabilities of HFEs in non-dominant sequences.	PERFORM detailed analyses for the estimation of the HEPs of the HFEs included in the model.	
HR-G2	USE an approach to estimation of HEPs that addresses failure in cognition as well as failure to execute		
HR-G3	USE an approach that takes the following into account: • the complexity of the response • the time available and time required to complete the response • 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: • quality (type (classroom or simulator) and fkquency) of the operator training or experience • quality of the written procedures and administrative controls • availability of instrumentation needed to take corrective actions • degree of clarity of the cues/indications • human-machine interface • Time Available and Time Required • complexity of the required response • environment (e.g. lighting, heat, radiation) under which the operator is working • accessibility of the equipment requiring manipulation • 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 quantification. 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:** • **the time required to complete all actions in** relation to the time **available to perform the actions** • **factors that could lead to dependence (e.g., common instrumentation, common procedures, increased stress, etc.)**		
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 4.4-5h. 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.

Index No. HR-H	CAPABILITY CATEGORY I	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III
HR-II1		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 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 • "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 4.4-5i 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.

Index No. HR-I	'CAPABILITY CATEGORY I	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III
IIR-11	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: • HRA methodology and process used to identify pre- and post-initiator HEPs • Generic and plant specific assumptions that were made in the HRA, including: ⇒ the bases for the assumptions ⇒ their impact on the CDF and LERF results • Factors used in the quantification of the human action, how they were derived (their bases), and how they were incorporated into the quantification process • •Source(s) of data used to quantify human actions, including: ⇒ screening values and their bases ⇒ best estimates with uncertainties and their bases ⇒ the method and treatment of dependencies for post-initiator actions ⇒ all pre- and post-initiator human actions evaluated by model, system, initiating event and function ⇒ all HEPs for each post-initiator human action and significant dependency effects		

4.4.6 DATA ANALYSIS

Objective: The objective of the'data analysis element is 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:

- parameters, whether estimated on the basis of plant specific or generic data, appropriately reflect the configuration and operation of the plant
- component or system unavailabilities due to maintenance or repair are accounted for
- uncertainties in the data are understood and appropriately accounted for

Table 4.4-6 HIGH LEVEL REQUIREMENTS FOR DATA ANALYSIS (HLR-DA)

A. Bach parameter shall be clearly defined in terms of the logic model, basic event boundary and the model used to evaluate event probability.
B. Grouping components into a homogeneous population for the purposes of parameter estimation shall consider both the design, environmental, and service conditions of the components in the as-built and as-operated plant.
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.
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.
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 4.4- 6a SUPPORTING REQUIREMENTS FOR DATA ANALYSIS HIGH LEVEL REQUIREMENT 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: • independent or common cause failure of a component or system to start or change state on demand • independent or common cause failure of a component or system to continue operating or provide a required function for a defined time period' • equipment unavailable to perform its required function due to being out of service for maintenance • equipment unavailable to perform its required function due to being in test mode • failure to recover a function or system (e.g. failure to recover offsite-power) • 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: • binomial distributions for failure on demand • Poisson distributions for standby and operating failures and initiating events		
DA-A3	IDENTIFY the parameter to be estimated and the data required. Examples include: • 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 demand • 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 4.4- 6b SUPPORTING REQUIREMENTS FOR DATA ANALYSIS HIGH LEVEL REQUIREMENT B

The **rationale** for grouping components into a homogeneous population for the purposes of 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 purposes of parameter estimation, GROUP components according to type (e.g., motor-operated pump, air-operated valve)	For purposes of parameter estimation, GROUP components according to type (e.g., motor-operated pump, air-operated valve) and according to the characteristics of their usage: • mission type (e.g., standby, operating), • service condition (e.g., clean vs. untreated water, air))	For purposes of parameter estimation, GROUP components according to type (e.g., motor-operated pump, air-operated valve) and according to the detailed characteristics of their usage: • design/size • system characteristics ⇒ mission type(e.g., standby, operating), ⇒ service condition(e.g., clean vs. untreated water, air) ⇒ maintenance practices ⇒ frequency of demands • environmental conditions • 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 valves that are never tested and unlikely to be operated with those that are tested or otherwise manipulated frequently) When warranted by sufficient data, USE appropriate hypothesis tests to ensure that data from grouped components are from compatible populations

Table 4.4- 6c SUPPORTING REQUIREMENTS FOR-DATA ANALYSIS HIGH LEVEL REQUIREMENT 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 SR-DA-A13. USE generic data for unavailability due to test, maintenance, and repair with caution since different plants can have different test and maintenance philosophi ies. Examples of parameter estimates and associated sources'include: • Component failure rates and probabilities – NUREG/ CR- 4639 [Reference 4.4.6- 1], NUREG/CR-4550 • Common cause failures – NUREG/ CR- 5497 [Reference 4.4.6- 2], NUREG/CR-6268 • AC off-site power recovery – NUREG/CR-5496, NUREG/CR-5032, NSAC-182/203 • Component recovery – NSAC-161		
DA-C2	COLLECT plant-specific data for the basic event/parameter grouping corresponding to that defined by requirement DA-A1.		
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: • 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). • INCLUDE as failures all events which 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.		

Table 4.4- 6c SUPPORTING REQUIREMENTS FOR DATA ANALYSIS HIGH LEVEL REQUIREMENT C
Generic parameter estimates shall be chosen and plant-Specific data shall be collected in accordance with the parameter defmitions 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-c!6	DETERMINE the number of plant-specific demands on standby components on the basis of the number of: • surveillance tests • maintenanceacts • surveillance tests or maintenance on other components. • 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 b&rational records to determine the time that components were configured in their standby status	
DA-C9	ESTIMATE operational time from surveillance test practices for standby components, and from actual operational data.		DETERMINE operational time from surveillance test records for standby components, and from actual operational data.

Table 4.4- 6c SUPPORTING REQUIREMENTS FOR DATA ANALYSIS HIGH LEVEL REQUIREMENT 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-C10	When using surveillance test data, REVIEW the test procedure to determine whether a test should be credited for each possible failure mode. Only COUNT complete 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. Only COUNT complete 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. Only COUNT complete 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.		

Table 4.4- 6c SUPPORTING REQUIREMENTS FOR DATA ANALYSIS HIGH LEVEL REQUIREMENT 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 IIII
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, only INCLUDE 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, only INCLUDE 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. CACULATE 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	Plant-specific data on recovery from loss of offsite power, loss of service water, etc. is 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 component failure until the component is returned to service.		

Table 4.4- 6d SUPPORTING REQUIREMENTS FOR DATA ANALYSIS HIGH LEVEL REQUIREMENT 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 exist with generic information for remaining events.	CALCULATE realistic parameter estimates for dominant contributors by using Bayesian updates. ADJUST prior distribution to account for plant-to-plant variability. CALCULATE parameter estimates for the remaining events using industry generic information.	CALCULATE realistic parameter estimates for dominant contributors by using Bayesian updates. ADJUST prior distribution to account for plant-to-plant variability. For cases where plant-specific data from a number of other plants is available, USE 2-stage Bayesian updating.
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 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. Example characterizations include: • Qualitative discussion • Sensitivity analyses	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.

Table 4.4- 6d SUPPORTING REQUIREMENTS FOR DATA ANALYSIS HIGH LEVEL REQUIREMENT D

The parameter estimates shall be based on relevant generic industiy 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-D4		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: • CONFIRM that the Bayesian updating does not produce a posterior distribution with a single bin histogram; • EXAMINE the cause of any unusual posterior distribution shapes • EXAMINE inconsistencies between the prior distribution and the plant- specific evidence to confirm that they are appropriate; • CONFIRM that the Bayesian updating algorithm provides valid results over the range of values being considered; • CONFIRM the reasonableness of the posterior distribution mean value.	
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 • Alpha Factor Model • Basic Parameter Model • Multiple Greek Letter Model • Binomial Failure Rate Model JUSTIFY the use of alternative methods.	USE one of the following models for estimating CCF parameters • Alpha Factor Model • Basic Parameter Model • Multiple Greek Letter Model • Binomial Failure Rate Model JUSTIFY the use of alternative methods.

Table 4.4- 6d SUPPORTING REQUIREMENTS FOR DATA ANALYSIS HIGH LEVEL REQUIREMENT 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)

ndex No.)A-D	CAPAIIILITY CATEGORY I	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III
)A-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 dominant common-cause events. An example approach is provided in NUREG/CR-5485. DETERMINE that the systems and data models are consistent.	
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: • 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; • 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: • 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; • 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 hiitori cal 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: • 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; • 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 thedatacanbeused.

Table 4.4- 6e SUPPORTING REQUIREMENTS FOR DATA ANALYSIS HIGH LEVEL REQUIREMENT 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-El	DOCUMENT the following: (a) system and component boundaries used to establi sh 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; and (j) parameter estimate including the characterization of uncertainty as appropriate.		

4.4.7 INTERNAL FLOODING

Objective:

The objective of the internal flooding element is 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:

- the water sources within the plant that could flood plant locations or create adverse conditions that could damage mitigative plant equipment are identified
- the flood scenarios/sequences that contribute to the core damage frequency and large early release frequency are identified and quantified.

TABLE 4.4-7 HIGH LEVEL REQUIREMENTS FOR INTERNAL FLOODING (HLR-IF)

A Different flood areas of the plant and the SSCs located within the areas shall be identified

B The potential flood sources in the plant and their associated flooding mechanisms shall be identified.

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.

D Flooding-induced initiating events shall be identified and their frequencies estimated.

E Flood-induced accident sequences shall be quantified.

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.

TABLE 4.4-7a

SUPPORTING REQUIREMENTS FOR INTERNAL FLOODING HIGH LEVEL REQUIREMENT 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:. □ the presence of physical barriers (e.g., walls, floors, dikes), □ mitigation features (e.g., sumps, drains), and □ 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: □ spatial information needed for the development of flood areas, □ SSCs located within each flood area, and □ potential flood sources within each flood area.		

TABLE 4.4-7b
SUPPORTING REQUIREMENTS FOR INTERNAL FLOODING HIGH LEVEL REQUIREMENT 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: . 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), . plant internal sources of water (e.g., tanks or pools) located in the area, and . 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: . failure modes of components such as pipes, tanks, gaskets, expansion joints, fittings, seals, etc.; . human-induced mechanisms that could lead to overfilling tanks, diversion of flow through openings created to perform maintenance; inadvertent actuation of fire suppression system; and . 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 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), . flow rate of water, and . capacity (e.g., gallons of water source).		
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 4.4-7c

SUPPORTING REQUIREMENTS **FOR INTERNAL FLOODING HIGH LEVEL REQUIREMENT 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 &Ives, 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.		

TABLE 4.4-7c

SUPPORTING REQUIREMENTS FORINTERNAL FLOODING HIGH LEVEL REQUIREMENT 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-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: • an area (including adjacent areas where flood sources can propagate) with no mitigating equipment modeled in the PRA; a flood within the area does not cause an initiating event, including a manual scram; or • an area with no significant flood sources (i.e., an area where there are no flood sources or where the volumes of the flood sources are insufficient to cause failure of equipment). • 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;		
IF-C6	USE potential human mitigative actions as additional criteria for screening if all the following can be shown: an area that has small or modest flood sources that is not in a propagation path from small or modest sources, the time to the damage of safe shutdown equipment is greater than 2 hours for the worst flooding initiator, flood indication is available in the control room, and 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: . an area that has small or modest flood sources that is not in a propagation path from small or modest sources, . the mitigative action can be performed with high reliability for the worst flooding initiator, . flood indication is available in the control room, and . 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 4.4-7d

SUPPORTING REQUIREMENTS FOR INTERNAL FLOODING HIGH LEVEL REQUIREMENT D

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

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Index No. IF-D	CAPABILITY CATEGORY I	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III
IF-D1	USE a structured, systematic process to identifying those flood scenarios 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	In searching for flood-induced initiating events, REVIEW the impact of plant-specific initiating event precursors and system alignments, INCLUDING alignments of supporting systems.		
IF-D3	GROUP flooding-induced initiating events only when: Events can be considered similar in terms of plant response, success criteria, and timing; or 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: 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 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: 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 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 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	For multi-unit sites with shared systems or structures, TREAT quantitatively dual unit internal flood initiators.	

TABLE 4.4-7d

SUPPORTING REQUIREMENTS FOR INTERNAL FLOODING HIGH LEVEL REQUIREMENT 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
	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.		
IF-D5	DETERMINE the flood initiating event frequency by using the applicable requirements in DA-C and one or more of the following: . an assessment of applicable generic operating experience of internal flooding, . an evaluation Of pipe, component, and tank rupture failure rates from generic data sources, . a probabilistic fracture mechanics evaluation of the probability of pipe leaks or ruptures representative of plant-specific conditions, . a combination of operating experience, and generic pipe and component failure rates, or . a combination of one of the above approaches with expert judgment.	DETERMINE the flood initiating event frequency by using the applicable requirements in DA-C and one or more of the following: • an assessment of generic and plant-specific operating experience of internal flooding, . an evaluation of pipe, component, and tank rupture failure rates from generic data sources enhanced by any plant-specific information, . a probabilistic fracture mechanics evaluation of the probability of pipe leaks or ruptures using plant-specific information, . a combination of operating experience and generic pipe and component failure rates, or . a combination of one of the above approaches with expert judgment.	

TABLE 4.4-7e

SUPPORTING REQUIREMENTS FOR INTERNAL FLOODING HIGH LEVEL REQUIREMENT E

Flood-induced accidents 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 Section 4.4.2 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 Section 4.4.4 to include flood-induced failures identified by IF-C3.		
IF-E4	PERFORM any additional data analysis to the applicable requirements described in Section 4.4.5.		
IF-E5	PERFORM any human reliability analysis to the applicable requirements described in Section 4.4.6 (HLR-HR-E through HLR-HR-G), and INCLUDE the following scenario specific PSFs for control room and ex-control room actions as appropriate: . additional workload and stress (above that for similar sequences not caused by internal floods), . uncertainties in event progression (e.g., cue availability and timing concerns caused by flood), and . effect of flood on mitigation, required response, and recovery activities (e.g., accessibility restrictions, possibility of physical harm) ❑ Iflooding-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 Section 4.4.8, 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 Section 4.4.9 and MODIFY as necessary to account for any flood-induced phenomena.		

TABLE 4.4-7f

SUPPORTING REQUIREMENTS FOR INTERNAL FLOODING HIGH LEVEL REQUIREMENT 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.		
IF-F2	DOCUMENT the following results: . flood sources identified in the analysis, any rules used to screen out these sources, and the resulting list of sources to be further examined; . flood areas used in the analysis and the reason for eliminating any of these areas from further analysis; . propagation pathways between flood areas and any assumptions, calculations, or other bases for eliminating or justifying any of these propagation pathways; . accident mitigating features and barriers credited in the analysis, the extent to which they were credited, and associated' justification; . 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; . screening criteria used in the analysis; . 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; . flood frequencies, component unreliabilities/unavailabilities, and HEPs used in the analysis (i.e., the data values unique to the flooding analysis); . any calculations or other analyses used to support or refine the flooding evaluation; and . 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.		

4.4.8 QUANTIFICATION

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

- The results reflect the design, operation, and maintenance of the plant
- Important contributors to CDF are identified in terms of initiating events, accident sequences, equipment failures, and operator errors
- Significant dependencies are accounted for
- Uncertainties are understood and appropriately quantified

Table 4.4-8 HIGH LEVEL REQUIREMENTS FOR THE LEVEL 1 ACCIDENT SEQUENCE QUANTIFICATION AND RESULTS INTERPRETATION

A. The level 1 quantification shall quantify core damage fkquency.
B. The quantification shall use appropriate models and codes, and shall account for method-specific limitations and features.
C. Model quantification shall determine that all identified dependencies are addressed appropriately.
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.
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.
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.

TABLE 4.4-8a

SUPPORTING REQUIREMENTS FOR QUANTIFICATION HIGH LEVEL REQUIREMENT 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 in order 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 4.4-8b

SUPPORTING REQUIREMENTS FOR QUANTIFICATION HIGH LEVEL REQUIREMENT B: The quantification shall use appropriate models and codes, and shall account for method-specific limitations and features. (HLR-QU-B)

Index No. Q U - B	CAPABILITY CATEGORY I	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III
QU-BI	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 in order 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-BS	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 (Reference [4.4.4-l]). When resolving circular logic, AVOID introducing significant 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 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-BS	When using logic flags, SET logic flag events to either TRUE or FALSE (instead of setting the event probabilities to 1 .O or O.O), as appropriate for each accident sequence, prior to the generation of cutsets.		
QU-B9	If modules, subtrees, or split tractions are used to facilitate the quantification, USE a process that allows: • Identification of shared events. • Correct formation of modules that are truly independent. • Results interpretation based on individual events within modules (e.g., risk significance).		

TABLE 4.4-8

SUPPORTING REQUIREMENTS **FOR QUANTIFICATION HIGH LEVEL REQUIREMENT C:**
Model quantification shall determine that all identified dependencies are addressed appropriately.
(HLR-QU-C)

Index No. Q U - 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.		

TABLE 4.4-8d

SUPPORTING REQUIREMENTS FOR QUANTIFICATION HIGH LEVEL REQUIREMENT D:

Thk 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		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.	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 4.4-8e

SUPPORTING REQUIREMENTS FOR QUANTIFICATION HIGH LEVEL REQUIREMENT 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.		
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.

TABLE 4.4-8e

SUPPORTING REQUIREMENTS FOR QUANTIFICATION HIGH LEVEL REQUIREMENT 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-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, HEPs, 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.

TABLE 4.4.8f

SUPPORTING REQUIREMENTS FOR QUANTIFICATION HIGH LEVEL REQUIREMENT 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 cut set 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 cut sets; (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 cut sets and their bases for elirninatibn		
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	

TABLE 4.4-8f

SUPPORTING REQUIREMENTS FOR QUANTIFICATION HIGH LEVEL REQUIREMENT 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-F3			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.		

4.4.9 LERF ANALYSIS

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

- The methodology is clear and consistent with the Level 1 evaluation, and creates an adequate transition from Level 1
- Significant operator actions, mitigation systems, and phenomena that can alter sequences are appropriately included in the LERF event tree structure and sequence definition
- Dependencies are reflected in the accident sequence model structure, if necessary
- 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
- End states are clearly defined to be LERF or non-LERF

Table 4.4-9 HIGH LEVEL REQUIREMENTS FOR LERF ANALYSIS (HLR-LE)

Plant Damage Analysis

A Core damage sequences shall be grouped into plant damage states based on their accident progression attributes

Accident Progression Analysis

B LERF evaluations shall include an analysis of the credible severe accident phenomena

C LERF evaluations shall include an analysis of containment system performance.

D LERF evaluations shall include an analysis of containmnent structural capability.

LERF Quantification

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

F LERF shall be quantified in a manner that captures factors important to risk and supports an understanding of the sources of uncertainty.

Documentation

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 4.4-9a

SUPPORTING REQUIREMENTS FOR LERF ANALYSIS HIGH LEVEL REQUIREMENT 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:*** • ***RCS pressure at core damage (high RCS pressure can result in High Pressure Melt Ejection)*** • ***status of emergency core coolant systems (failure in injection can result in a dry cavity and extensive Core Concrete Interaction)*** • ***status of containment isolation (failure of isolation can result in an unscrubbed release)*** ***status of containment heat removal***		
LE-A2	IDENTIFY the accident sequence **characteristics** that lead to the **physical characteristics identified** in LE-Al . ***Examples include:*** • ***Type of initiator*** ⇒ ***Transients can result in high RCS pressure,*** ⇒ ***LOCAs usually result in lower RCS pressure,*** ⇒ ***ISLOCAs, SGTRs can result in containment bypass.*** • ***status of electric power*** • ***loss of electric power can result in loss of ECC injection*** • ***status of containment safety systems such as sprays, fan coolers, igniters, or venting systems*** • ***operability of containment safety systems determines status of containment heat removal*** ***References [4.4-9-1] and [4.4-9-2] 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-Al and **LE-A2** characteristics and ensure that dependencies are properly treated.		
LEAS	DEFINE plant damage states based upon information in **AS-A8**, LE-Al, **LE-A2**, **LE-A3**, and **LE-A4**..		

[References:

[4.4-9-1] Nuclear Power Plant Response to. Severe Accident IDCOR Technical **Summary** Report, **Atomic** Industrial Forum, November, 1984

[4.4-9-2] NUREG 11 **50,"Severe** Accident Risks: An Assessment for Five U.S. Nuclear Power Plants", USNRC, December , 1990

TABLE 4.4-9b

SUPPORTING REQUIREMENTS FOR LERF ANALYSIS HIGH LEVEL REQUIREMENT B

. LERF ANALYSIS **(HLR-LE-B):** LERF **evaluations** shall **include an analysis of** the credible severe accident phenomena.

Index No. LE-B	CAPABILITY CATEGORY I	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III

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TABLE 4.4-9b

SUPPORTING REQUIREMENTS FOR LERF ANALYSIS HIGH LEVEL REQUIREMENT B

LERF ANALYSIS (HLR-LE-B): LERF evaluations shall include an analysis of the credible severe accident phenomena.

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 4.4-9(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 NUREGICR-6595. • INCLUDE as appropriate, unique plant issues as determined by expert judgement and/or past plant analyses. • EVALUATE qualitatively those severe accident phenomena that are not quantified in NUREG/CR-6595 (e.g., induced steam generator tube rupture)	INCLUDE potential severe accident phenomena that could impact LERF from the set identified in Table 4.4-9(a). This is the minimum set to be considered. INCLUDE as appropriate, unique plant issues as determined by expert judgement 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 or in NUREG-1150. 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. 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.

TABLE 4.4-9b

SUPPORTING REQUIREMENTS FOR LERF ANALYSIS HIGH LEVEL REQUIREMENT B

LERF ANALYSIS (HLR-LE-B): LERF evaluations shall include an analysis of the credible severe accident phenomena.

Index No: LE-B	CAPABILITY CATEGORY I	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III
LE-B3	USE available containment analyses from generic or plant specific sources. An acceptable alternative is the approach in NUREG/CR-6595.	USE plant specific containment thermal hydraulic analyses to model containment and RPV/RCS response under severe accident progression.	

Table 4.4-9(a): Potential LERF Contributors to be Considered

Potential LERF Contributor	Containment Desi				
	Large Dry and Subatmospheric	Ice Condenser	BWR Mark I	BWR Mark II	BWR Mark III
Containment Isolation Failure	✓	✓	✓	✓	✓ (1)
Containment Bypass					
- ISLOCA	✓	✓	✓	✓	✓
- SGTR	✓	✓			
- Induced SGTR	✓	✓			
Energetic Containment Failures					
- HPME	✓	✓	✓	✓	✓
-Hydrogen Combustion		✓			✓
- De-inerted Operation			✓	✓	
-Core Debris Impingement	(2)	✓	✓	✓	
Steam Explosion			✓	✓	✓
Shell Melt-through			✓		
Suppression Pool Bypass			✓		✓
RPV and/or Containment Venting			✓	✓	✓
Isolation Condenser Tube Rupture			✓ (if applicable)		
Vacuum breaker failure			✓	✓	✓
Hydrodynamic Loads under severe accident conditions			✓		✓
Containment flooding			✓		
In-vessel recovery	✓	✓	✓	✓	✓
ATWS-Induced Failure			✓	✓	✓

(1) Dry Well (DW) Isolation Failure
(2) Steel shell designs

NOTE: Combinations of phenomena should also be considered where appropriate

TABLE 4.4-9c

SUPPORTING REQUIREMENTS FOR LERF ANALYSIS HIGH LEVEL REQUIREMENT 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** (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.** USE **HFEs** consistent with **EOPs/Severe** Accident **Management** Guidelines **(SAMGs),** proceduralized actions or Technical Support Center guidance.	**INCLUDE realistic treatment of feasible** operator actions follo**wing the onset of core damage** **USE HFEs consistent with EOPs/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.

TABLE 4.4-9c
SUPPORTING REQUIREMENTS FOR LERF ANALYSIS HIGH LEVEL REQUIREMENT 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-C3	INCLUDE those branch points necessary to provide a conservative LERF estimation. Containment event trees developed in **NUREG/CR-6595** (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.
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 which support **LERF consistent with the applicable requirements** of Table 4.4-4, as appropriate for the level of detail of the analysis.		
LE-C6	DEFINE **HFEs** which support LERF consistent with the applicable requirements **of Table** 4.4-5, 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 Table 4.4-2, as appropriate for the **level** of detail of the **analysis.**		
LE-C8	TREAT containment environmental impacts on continued operation of equipment **&** operator actions in a conservative manner. An acceptable alternative is the approach in NUREG/CR-6595. A realistic treatment **may be used.**	TREAT containment **environmental** impacts on **continued** operation of equipment **&** 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 **&** operator actions in a realistic manner when possible.

TABLE 4.4-9c

SUPPORTING REQUIREMENTS FOR LERF ANAL-YSIS HIGH LEVEL REQUIREMENT C

LERF evaluations shall include an analysis of containmentsystem performance **(HLR-LE-C)**

Index No. LE-C	CAPABILITY CATEGORY I	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III
LE-C9	TREAT containment failure impacts on continued operation of equipment & operator actions in conservative manner. An acceptable alternative is the approach in NUREGICR-6595. A realistic treatment may be used.	**TREAT containment failure impacts on** continued operation of equipment & operator actions in a realistic manner **when possible. Conservative treatment may be used for non-dominant LERF contributors, if their use does not distort insights.**	TREAT containment failure impacts on continued operation of equipment & operator actions in a realistic manner when possible.
LE-Cl0	TREAT containment bypass events in a conservative manner. DO NOT TARE CREDIT for reducing the class of the release by considering scrubbing. An acceptable treatment of containment bypass is in **NUREG/CR-6595.** 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**	TREAT containment bypass events in a realistic manner.

TABLE 4.4-9d
SUPPORTING REQUIREMENTS FOR LERF ANALYSIS HIGH LEVEL REQUIREMENT 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. Quasi-static containment capability evalwtions 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 i n **NUREG/CR-6595.**	**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.** **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 : • containment seals, • penetrations, • hatches • **drywell** head **(BWRs)** • **and** vent pipe bellows **(BWRs)** beyond the design basis temperature and pressure conditions. PROVIDE. static and dynamic failure capabilities, as appropriate.

TABLE 4.4-9d
SUPPORTING REQUIREMENTS FOR LERF ANALYSIS HIGH LEVEL REQUIREMENT 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
LED2	When failure location [1] 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.	When failure location [1] 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 a realistic 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 realistic 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.
LED5	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..
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 which 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.

(1) 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 4.4-9e

SUPPORTING REQUIREMENTS FOR LEVEL 2 QUANTIFICATION HIGH LEVEL REOUIREMENT 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	CAPABTLTTY CATEGORY III
LE-El	SELECT parameter values used in LERF analysis consistent with the applicable requirements of Tables 4.4-5 and 4.4-6 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.**.	**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 Table 4.4-8, as appropriate for the level of detail of the analysis.		

TABLE 4.4-9f

SUPPORTING REQUIREMENTS FOR LEVEL 2 QUANTIFICATION HIGH LEVEL REQUIREMENT 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: • Identify bounding assumptions, • 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 4.4-9g

SUPPORTING' REQUIREMENTS FOR LEVEL 2 QUANTIFICATION HIGH LEVEL REQUIREMENT 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-Gl	DOCUMENT: • The physical characteristics and accident sequence characteristics used to group the accident sequences into plant damage states • The method used to bin the accident sequences into plant damage states • The plant damage states and their attrbutes, as used in the analysis.		
LE-G2	DOCUMENT the potential LERF contributors considered, where appropriate, including: • containment isolation failure • containment bypass • energetic containment failures • hydrogen related phenomena • liner melt-through (Mark I containment) • containment failure at vessel breach • induced SGTRs (PWRs) • containment venting . • suppression pool bypass mechanism (BWRs) • containment flooding (BWRs)		
LE-G3	DOCUMENT treatment of key factors influencing containment capability, as appropriate for the level of 'detail of the analysis, including: • design details (i.e., heat sink distribution, circulation paths, ignition sources, water availability, and gravity drain paths, cavity geometry) • equipment survivability credited beyond design basis envelope		

TABLE 4.4-9g

'SUPPORTING REQUIREMENTS FOR LEVEL 2 QUANTIFICATION HIGH LEVEL REQUIREMENT G

The documentation of the LERF analysis shall be perfonned 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-GS	DOCUMENT, as appropriate for the level of detail of the analysis: • containment challenges treated • containment failure modes identified • containment Event Tree (CET) and basis for event tree nodes • containment capacity and its basis • containment failure locations and probabilities • basis for parameter estimates		
LE-G6	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 general description of the quantification • important assumptions • the total plant LERF and contributions from the different plant damage states and accident classes; • equipment or human actions that are the key factors; • the results of all sensitivity studies, (as applicable).		
LE-G7	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-G8			DOCUMENT sources of uncertainty.
LE-G9	IDENTIFY limitations that would impact applications.		

5 PRA CONFIGURATION CONTROL

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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 process for monitoring PRA inputs and collecting new information
- a process that maintains and upgrades the PRA to be consistent with the as-built as operated plant
- a process that ensures that the cumulative impact of pending changes is considered when applying the PRA
- a process that evaluates the impact of changes on previously implemented risk-informed decisions that have used the PRA
- a process that maintains configuration control of computer codes used to support PRA quantification
- 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 Section 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 Supporting Requirements pertinent to that application as determined through the process described in Section 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 shah 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.

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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.

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 consistent with the as-built, **as**-operated plant.

The documentation typically includes:

- a **description** of the process used to monitor PRA inputs and collect new information
- evidence that the aforementioned process is active
- descriptions of proposed changes
- description of changes in a **PRA** due to each PRA upgrade or PRA maintenance
- record of the performance and results of the appropriate PRA reviews
- record of the process and results used to address the cumulative impact of pending changes
- record of the process and results used to evaluate changes on previously implemented risk-informed decisions pursuant to subsection 5.6
- a description of the process used to maintain software configuration control

6 PEER REVIEW

Contents

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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 against the Elements contained in Section 4 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 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; and

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

Reference [6.1.2-1] provides an example of an acceptable review methodology; however, the differences between the supporting requirements of Section 4.0 of this Standard and the supporting requirements of Appendix B, Reference 6.1.2-1 shall be evaluated.

References

[6.1.2-l] Probabilistic Risk Assessment (PRA) Peer Review Process Guidance, NEI-00-02.

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; and

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

6.2.2 General. The peer review team members individually shall:

(a) be knowledgeable of the requirements in this Standard for their area of review; and

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

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, and

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

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 subsection may be taken based on the availability of. appropriate personnel to develop a team. All such exceptions shall be documented in accordance with Subsection 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 paragraphs of 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 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 High Level Requirements and the composite of the supporting Requirements of Section 4 shall be used by the peer review team to assess the completeness of a PRA Element.

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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:

- accident sequence model for a balance-of-plant transient
- the accident sequence model containing LOOP/Station Blackout considerations
- accident sequence model for a loss of a support system initiating event
- **LOCA** accident sequence model
- **ISLOCA** accident sequence model
- the SGTR accident sequence model (for **PWRs** only)
- **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 ffor review typically includes:

- the definition of core damage used in the success criteria evaluations and the supporting bases
- the modeling of conditions corresponding to a safe stable state
- the core and containment response' conditions used in defining **LERF** and supporting bases
- the core and containment system success criteria used in the PRA for mitigating each modeled initiating event
- the generic bases (including assumptions) used to establish the success criteria of systems credited in the PRA and the applicability to the modeled plant
- the plant-specific bases (including assumptions) used to establish the system success criteria of systems credited in the PRA
- calculations **performed** specifically for the **PRA,** for each computer code used to establish core cooling or decay heat removal success criteria and accident sequence timing
- 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 a SBO)
- the containment **response** calculations, performed specifically for the PRA, for the dominant plant damage states
- **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:

- dominant systems contributing to the CDF or LERF calculated in the PRA,
- different models reflecting different levels of detail
- front-line system for each mitigating function (e.g., reactivity control, coolant injection, and decay heat removal)
- each major type of support system, (e.g.,, electrical power, cooling water, instrument air, and HVAC)
- coniplex 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 (HRA). A review shall be performed on the human reliability analysis.

The portion of the HRA selected for review typically includes:

- HEPs for dominant human actions contributing to the CDF or LERF calculated in the PM
- the selection and implementation of any screening HEPs used in the PRA
- post-accident HEPs
- pre-initiator HEPS for both instrumentation misc.alibration and failure of equipment
- HEPs for the same human action but with different times required for success
- HEPs for dependent human actions.
- HEPs less than 1E-4
- 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:

- data values for component failure modes contributing to the CDF or LERF calculated in the PRA
- common cause failure values
- 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)
- 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:

- dominant internal flooding contributors to the CDF or LERF calculated in the PRA
- the screening of any flood areas
- internal flood initiating event frequencies
- internal flooding scenario involving each identified flood source
- internal flooding scenarios involving flood propagation to adjacent flood areas
- internal flooding scenario that involves each of the flood-induced component failure mechanisms (i.e., one flood scenario for each mechanism)
- 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:

- appropriateness of the computer codes used in the quantification
- the truncation values and process

- the recovery analysis
- model asymmetries and sensitivity studies
- the process for generating modules (if used)
- logic flags (if used)
- 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.

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

- accident characteristics chosen for carryover to LERF analysis (and for binning of PDS if PDS methods were used)
- interface mechanism used and
- SDE carryover

The portion of the LERF analysis selected :for review typically includes:

- . the **LERF** analysis method;.
- demonstration that the phenomena that impact radionuclide release characterization of **LERF** have been appropriately considered .
- human action and system success considering adverse conditions that would exist following core damage,
- the sequence mapping'
- evaluation of containment performance under severe accident conditions
- the definition and bases for **LERF**
- inclusion in the containment event tree of the functional events necessary to achieve a safe stable . containment endstate

6.4 Expert Judgment

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

6.5 **PRA Configuration Control**

The pee; 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 of exceptions and deficiencies with respect to Supporting Requirements and their

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

(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.

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