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Treatment of Parameter and Modeling Uncertainty for Probabilistic Risk Assessments

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REPORT SUMMARY

Background

Both the industry and the U.S. Nuclear Regulatory Commission (NRC) incorporate risk concepts and techniques into activities for effective risk management. The NRC is using probabilistic risk assessment (PRA) in its regulatory activities in a manner that promotes consistency, predictability, and efficiency in the performance of the NRC's roles of risk manager and protector of public health and safety. The nuclear industry uses PRA to identify and manage risks, as a tool to promote efficient regulatory interaction, and to increase operational flexibility.

The characterization of uncertainty is desirable because it supports the effective, informed use of PRA results. As such, one of the main purposes of uncertainty assessment in probabilistic models is to facilitate the decision-making process. Another is to aid in the refinement of the models themselves in order to reduce uncertainty where possible.

Objectives

- To provide technical guidance that establishes a level of confidence in a decision about or a conclusion based on a quantitative assessment of risk. The quantitative assessment of risk involves doing the following:
  - Identifying when point estimate solutions are not suitable in light of parametric uncertainties
  - Generically defining those modeling uncertainty issues that need to be identified and characterized in order to meet the supporting requirements of selected American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA standards
  - Providing an acceptable method for meeting the requirements of Regulatory Guide (RG) 1.200 related to identifying and characterizing sources of modeling uncertainty in the base PRA
  - Providing an acceptable method for evaluating the impact of uncertainties in the application of quantitative acceptance guidelines that are part of the NRC's risk-informed regulatory processes
- To complement the NRC report *Guidelines on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making* (NUREG-1855). Together, the two reports provide a single approach to the treatment of uncertainty for the base model and risk-informed applications.

Approach

The research contained is this report is a culmination of several efforts, including *Guideline for the Treatment of Uncertainty in Risk-Informed Applications: Technical Basis Document* (EPRI report 1009652), the application guide with the same title (EPRI report 1013491), pilot efforts by Exelon and Electricité de France, and NUREG-1855.

Results

Section 2 presents guidance on parametric uncertainty characterization for use in meeting the ASME/ANS PRA standard. In particular, the guidance addresses state-of-knowledge correlation (SOKC) when evaluating the risk metrics under PRA. Note that NUREG-1855 refers to SOKC as *epistemic correlation*.

Section 3 describes how a previously developed, long list of potential sources of uncertainty was reviewed, leading to the earmarking of certain issues as candidates for modeling uncertainty. Now that the narrower list of candidates exists, a plant-specific issue characterization for the base PRA model can be provided in order to meet the relevant supporting requirements from the standard for the base PRA model. This guidance also appears in Section 3 of this report.

Section 4 provides guidance on characterizing modeling uncertainties in the context of risk-informed applications. A framework for the selection, preparation, assessment, and reporting of results of sensitivity studies to account for uncertainties in the context of decision making is laid out.

Appendix A is a list of those issues that were identified as candidate sources of modeling uncertainty, and it provides additional complementary information to meet the applicable ASME/ANS PRA standard supporting requirements listed in Section 1.3.1 of this report. Finally, Appendix B describes a sample implementation of the process for the base model assessment described in Section 3.

EPRI Perspective

This report provides an overall framework for the pragmatic treatment of uncertainty characterization that is to be used in risk-informed decision making. It is considered an industry good practice, and it uses the most current techniques and information. The main purpose of assessing uncertainty can be restated as the following: to provide reasonable assurance that the risk-informed decision, made based on comparisons to specific acceptance guidelines, is not unduly influenced by uncertainties in the PRA results; it does not, therefore, warrant reconsideration.

Keywords

American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS)
    PRA standard
Consensus model
Probabilistic risk assessment (PRA)
State-of-knowledge correlation (SOKC)
Uncertainty

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1 INTRODUCTION

1.1 Purpose and Scope

The purpose of this report is to provide guidance for a structured process for addressing uncertainties in probabilistic risk assessment results in the context of risk-informed decision-making. It is a follow-on to the EPRI report, *Guidelines for the Treatment of Uncertainty in Risk-Informed Applications: Technical Basis Document* (1009652), which was published in 2004 [1] and it supersedes the companion *Applications Guide* (1013491) which was published in 2006 [2]. Additionally, the intent of this document is to complement the NRC report, *Guidelines on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making* (NUREG-1855) [3], anticipated to be released near the end of 2008. This effort has been undertaken to provide the following benefits to the industry and the NRC in moving forward with risk-informed applications:

- Identify when point estimate solutions are not suitable in light of parametric uncertainties
- Assist utilities in identifying and characterizing sources of model uncertainty
- Provide guidance for identifying appropriate sensitivity cases and logical combinations of sensitivity cases
- Provide guidance for identifying key assumptions associated with risk informed applications

The intent is to provide a pragmatic process for uncertainty characterization that is to be used in risk-informed applications and decision making. It is considered an industry good practice, and it uses state-of-the-technology information. The main purpose of assessing uncertainty is to establish the level of confidence that can be placed in a decision or conclusion based on a quantitative assessment of risk. That is, the main purpose is to provide reasonable assurance that the risk-informed decision, made based on guidelines either specific to the application or as part of NRC Regulatory Guide (RG) 1.174 [4], is appropriate given the uncertainties in the PRA results and would therefore not warrant reconsideration.

The scope of the uncertainty evaluation guidelines development is limited to that supporting the ASME/ANS PRA Standard [26] for at-power internal events for the current fleet of operating reactors. As a starting point, the base PRA being considered for the application of these uncertainty guidelines is assumed to be developed and quantified largely consistent with

Capability Category II of the ASME/ANS PRA Standard. For reference purposes, Section 1.1.3 (Table 1.1.3-2) of the ASME/ANS PRA Standard provides a useful description for the bases of the PRA capability categories. For Capability Category II, the definitions provided below apply.

- Scope and level of detail: PRA model resolution and specificity sufficient to identify the relative importance of the significant contributors at the component level including associated human actions, as necessary.
- Plant-specificity: Use of plant-specific data/models for the significant contributors.
- Realism: Departures from realism will have small impact on the conclusions and risk insights as supported by good practices.

1.2 Background

The safe, economical operation of a nuclear power plant is accomplished through effective management review and control, including risk management. Risk management can be segmented into a number of different facets. The design, procurement, construction, operation, and maintenance of a nuclear power plant are accomplished through a variety of programs, procedures, and administrative controls, many of which are subject to regulatory requirements aimed at ensuring adequate protection of the public health and safety. One of these perspectives is currently shaping the way risk management is practiced at some utilities: the use of PRA insights. With the advancements in PRA technology, PRA insights are being used with existing deterministic considerations in assessing plant activities and as inputs to the decision-making processes.

Since the early 1990s and the publishing of NUREG-1150, *Severe Accident Risks: An Assessment of Five U.S. Nuclear Power Plants,* the NRC has increased its use of risk-informed concepts [7]. The NRC has used risk-informed concepts in a number of regulatory policies and programs, notably:

- Safety goal policy statement
- Individual plant examination program
- Maintenance Rule

In many of these policy statements and programs, the ability to effectively deal with uncertainties is identified as part of the process. The treatment of uncertainties can many times consist of recognition of the uncertainties and the acknowledgment that the decisions are made based on the realistic, best-estimate values from the probabilistic models coupled with margin designed into the acceptance guidelines and with the defense-in-depth deterministic inputs.

Neither the NRC nor the industry intends to use PRA as a replacement for traditional deterministic approaches. PRA is viewed as a complement to the deterministic method. In fact, probabilistic and deterministic methods are acknowledged as extensions of each other rather than as separate and distinct.

Both the industry and the NRC are incorporating risk concepts and techniques into activities for effective risk management. The NRC is using PRA in regulatory activities in a manner that promotes consistency, predictability, and efficiency in the performance of its role of risk management and protection of public health and safety. The nuclear industry is using PRA to identify and manage risks as a tool to promote efficient regulatory interaction and to increase operational flexibility.

The NRC has provided high-level guidance on the treatment of parameter and model uncertainties in two regulatory guidelines issued to support risk-informed regulatory decisions:

- RG 1.174 Guidance on Addressing Parameter Uncertainties [4]
  - Section 2.2.5.5 states:

    "Because of the way the acceptance guidelines were developed, the appropriate numerical measures to use in the initial comparison of the PRA results to the acceptance guidelines are mean values. The mean values referred to are the means of the probability distributions that result from the propagation of the uncertainties on the input parameters and those model uncertainties explicitly represented in the model. While a formal propagation of the uncertainty is the best way to correctly account for state-of-knowledge uncertainties that arise from the use of the same parameter values for several basic event probability models, under certain circumstances, a formal propagation of uncertainty may not be required if it can be demonstrated that the state-of-knowledge correlation is unimportant. This will involve, for example, a demonstration that the bulk of the contributing scenarios (cutsets or accident sequences) do no involve multiple events that rely on the same parameter for their quantification."
- RG 1.174 Guidance on Addressing Model Uncertainties [4]
  - Section 2.2.5.5 also states:

    "Whether the PRA is full scope or only partial scope… it will be incumbent on the licensee to demonstrate that the choice of reasonable alternative hypotheses, adjustment factors, or modeling approximations or methods to those adopted in the PRA model would not significantly change the assessment. This demonstration can take the form of well formulated sensitivity studies or qualitative arguments. In this context, "reasonable" is interpreted as implying some precedent for the alternative, such as use by other analysts, and also that there is a physically reasonable basis for the alternative. It is not the intent that the search for alternatives should be exhaustive and arbitrary."
- RG 1.200 Guidance on Addressing Parameter Uncertainties [6]
  - Section 1.2.6 states:

    "Parameter estimation analysis quantifies the frequencies of initiating events, as well as the equipment failure probabilities and equipment unavailabilities of the modeled systems. The estimation process includes a mechanism for addressing uncertainties and has the ability to combine different sources of data in a coherent manner, including the actual operating history and experience of the plant when it is of sufficient quality, as well as applicable generic experience."

- RG 1.200 Guidance on Addressing Model Uncertainties [6]
  - Section 1.2.6 also states:

    "An important aspect in understanding the PRA results is understanding the associated uncertainties. Sources of uncertainty are identified and their impact on the results analyzed. The potential conservatism associated with the successive screening approach used for the analysis of specific scope items such as fire, flooding, or seismic initiating events is assessed. The sensitivity of the model results to model boundary conditions and other assumptions is evaluated using sensitivity analyses to look at assumptions both individually or in logical combinations. The combinations analyzed are chosen to account for interactions among the variables."

Because the *uncertainty evaluation* is intimately tied to the *decision under consideration*, the quantitative measurements of uncertainty and ultimately the acceptance of the decision must reflect the relationship between the two. There can be considered two principal distinctions in traditional PRA analysis when considering the treatment of uncertainty. The first is the base model PRA where uncertainties may influence the primary risk metric such as core damage frequency (CDF) or Large Early Release Frequency (LERF) and secondary risk metrics such as the functional accident sequence group frequencies. The second is the application of the PRA where other risk metrics such as delta core damage frequency (ΔCDF) are used and where the degree of uncertainty in specific areas of the PRA model may become more or less important depending on the application.

As the use of risk technology has matured and additional applications of that technology are pursued, there is a desire to develop a process for the treatment of uncertainty that is also consistent with the intent of the ASME/ANS standard. The ASME/ANS standard and RG-1.200 require that baseline PRA uncertainties be characterized. The appropriate characterization of the uncertainty is a desirable goal to support the effective and informed use of the PRA results. A secondary benefit of this process is that knowledge of uncertainty provides an opportunity to aid in the refinement of the models themselves. These refinements to the PRA model usually take the form of additional analyses to reduce the uncertainty or conservatism in a portion of the model identified by the treatment of uncertainty.

The methodology for treatment of uncertainties considered in this report (and its companion document NUREG-1855) is intended to provide a reasonable process to satisfy specific ASME/ANS supporting requirements. It should be noted that the treatment of uncertainty methodology contained in this report does not supersede regulatory accepted treatments for specific applications.

1.3 Overview

The order in which type of uncertainty are presented and the guidance developed is consistent with the literature regarding uncertainty treatment, for example RG 1.174 [4] where the three high-level types of uncertainties are the following:

- Parametric uncertainty
- Modeling uncertainty
- Completeness uncertainty

The first released draft of NUREG-1855 provided useful definitions for these three sources of uncertainty.

<u>Parameter Uncertainty</u>

> "Parameter uncertainty relates to the uncertainty in the computation of the parameter values for initiating event frequencies, component failure probabilities, and human error probabilities that are used in the quantification process of the PRA model. These uncertainties can be characterized by probability distributions that relate the analysts' degree of belief in the values that these parameters could take. Most of the PRA software in current use has the capability to propagate these uncertainties through the analysis and calculate the probability distributions for the results of the PRA. To make a risk-informed decision, the numerical results of the PRA, including their associated uncertainty, must be compared with the appropriate decision guidelines."

<u>Model Uncertainty</u>

> "Model uncertainty relates to the uncertainty in the assumptions made in the analysis and the models used. Examples of model uncertainty include the assumptions made as to how a reactor coolant pump in a pressurized water reactor would fail following loss of seal cooling and/or injection, the approach used to address common cause failures in the PRA model, and the approach used to identify and quantify operator errors. In general, model uncertainties are addressed by studies to determine the sensitivity of the results of the analysis if different assumptions are made or different models are used."

<u>Completeness Uncertainty</u>

> "Completeness uncertainty […] relates to contributions to risk that have been excluded from the PRA model. This class of uncertainties may have a significant impact on the predictions of the PRA model and must be addressed. Examples of sources of incompleteness include the following:
>
> - The scope of the PRA does not include some class of initiating events, hazards, or modes of operation.
> - There is no agreement on how the PRA should address certain elements, such as the effects on risk resulting from aging or organizational factors.
> - The analysis may have omitted phenomena, failure mechanisms, or other factors because their relative contribution is believed to be negligible."

Parameter uncertainty and model uncertainty will be specifically addressed in the context of this report. The treatment of completeness uncertainty will not be specifically addressed here. However, risk significant scope issues should be understood and considered in the context of the application, as applicable. Note that the treatment of completeness uncertainty is more fully addressed in NUREG-1855.

1.3.1 Applicable ASME/ANS PRA Standard Supporting Requirements

For parameter uncertainty, two supporting requirements of the ASME/ANS PRA Standard [26] specifically address the treatment of the state-of-knowledge correlation. (Note that the "state-of-knowledge" correlation is referred to as the "epistemic" correlation in NUREG-1855.) The Capability Category II definitions for these two supporting requirements (SRs), QU-A3 and QU-E3 are shown below. Two additional SRs (LE-E4 and LE-F3) ensure that the same requirements exist for LERF as they do for CDF.

- QU-A3: ESTIMATE the mean CDF accounting for the "state-of-knowledge" correlation between event probabilities when significant.
- QU-E3: ESTIMATE the uncertainty interval of the CDF results. ESTIMATE the uncertainty intervals associated with parameter uncertainties (DA-D3, HR-D6, HR-G8, IE-C15) taking into account the "state-of-knowledge" correlation.

The generic sources of model uncertainty have been identified based on interactions with NRC and industry personnel. The list has been developed with the intent of defining that set of issues that need to be addressed to satisfy the following supporting requirements from the ASME/ANS PRA Standard.

- QU-E1: IDENTIFY sources of model uncertainty.
- QU-E2: IDENTIFY assumptions made in the development of the PRA model.
- QU-E4: For each source of model uncertainty and related assumption identified in QU-E1 and QU-E2, respectively, IDENTIFY how the PRA model is affected (e.g., introduction of a new basic event, changes to basic event probabilities, change in success criterion, introduction of a new initiating event).
- QU-F4: DOCUMENT the characterization of the sources of model uncertainty and related assumptions (as identified in QU-E4).
- LE-F3: IDENTIFY and CHARACTERIZE the LERF sources of model uncertainty and related assumptions, consistent with the requirements of Tables 2.2.7-2(d) and 2.2.7-2(e).
- IE-D3, AS-C3, SC-C3, SY-C3, HR-I3, DA-E3, LE-G4, and IF-B3: DOCUMENT the sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2 [or LE-F3]) associated with … [each element].

1.3.2 Supporting Definitions

In the process of identifying and characterizing assumptions associated with the potential sources of modeling uncertainty, it is important to establish a definition for such assumptions. These definitions have also been provided in the ASME/ANS PRA Standard [26].

An *assumption* is a decision or judgment that is made in the development of the PRA model. An assumption is either related to a source of model uncertainty or is related to scope or level of detail.

An *assumption related to a model uncertainty* is either generally accepted, or is made with the knowledge that a different reasonable alternative assumption exists. *A reasonable alternative assumption* is one that has broad acceptance within the technical community and for which the technical basis for consideration is at least as sound as that of the assumption being made.

Example 1: Detailed analyses demonstrate that temperatures in a region would be less than required to fail a component. An alternative recommendation to consider the component failed is offered without strong basis. This additional analysis is not required to satisfy the ASME/ANS or RG 1.200 requirement.

Example 2: An assumption that a component fails when exposed to an unspecified harsh environment has a probability of failure of 0.50. This assumption is important to a specific application. An alternative recommendation that the probability be increased to 1.0 (always fails) is presented. If there is no clear basis for the initial value, an impact of the full failure condition should be considered.

An *assumption related to scope or level of detail* is one that is made for modeling convenience.

Example 1: Single basic events are utilized to represent the mechanical or electrical failures of RPS. A detailed representation of the RPS system is not necessary for most applications.

Example 2: All failures to run for the Low Pressure Core Injection (LPCI) / Residual Heat Removal (RHR) system are assumed to occur at time zero. This conservatism is generally acceptable for most applications.

Note that the definitions above provide delineation for, or guidance for the identification of, those sources of model uncertainty (and related assumptions) that should be the focus for meeting the QU supporting requirements described above. These definitions focus on those assumptions or model treatment that are truly related to uncertainty as opposed to assumptions and model treatments or developments that are related to scope or level of detail.

Also note that the types of reasonable alternative assumptions related to sources of model uncertainty can lead to increases or decreases in the calculated risk metrics, and the process should recognize that both outcomes are possible. The development of the base PRA model therefore requires a careful balance of conservative bias treatment with realistic assessments to ensure that the base PRA model risk profile is adequately developed. Applications of the PRA model could become more or less dependent on the choices made for addressing modeling uncertainty.

1.4 Objective

The objective of this report is to provide technical guidance for establishing the level of confidence that can be placed in a decision or conclusion based on a quantitative assessment of risk by:

1. Identifying when point estimate solutions are not suitable in light of parametric uncertainties
2. Defining, generically, those set of modeling uncertainty issues that need to be identified and characterized to meet the supporting requirement definitions provided above
3. Providing an acceptable method for meeting the requirements of RG 1.200 related to identifying and characterizing sources of modeling uncertainty in the base PRA
4. Providing an acceptable method for evaluating the impact of uncertainties in the application of quantitative acceptance guidelines that are part of the NRC's risk-informed regulatory processes

Additionally, the intent of this document is to complement the NRC report, *Guidelines on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making* (NUREG-1855). An overview of the relationship between these two documents is shown in Figure 1-1. In general, the process described here is consistent with a portion of the overall structure and framework for risk-informed decision-making as outlined in NUREG-1855, and provides additional details in pertinent places. As stated previously, the treatment of completeness uncertainty will not be specifically addressed here, but it is addressed in NUREG-1855.

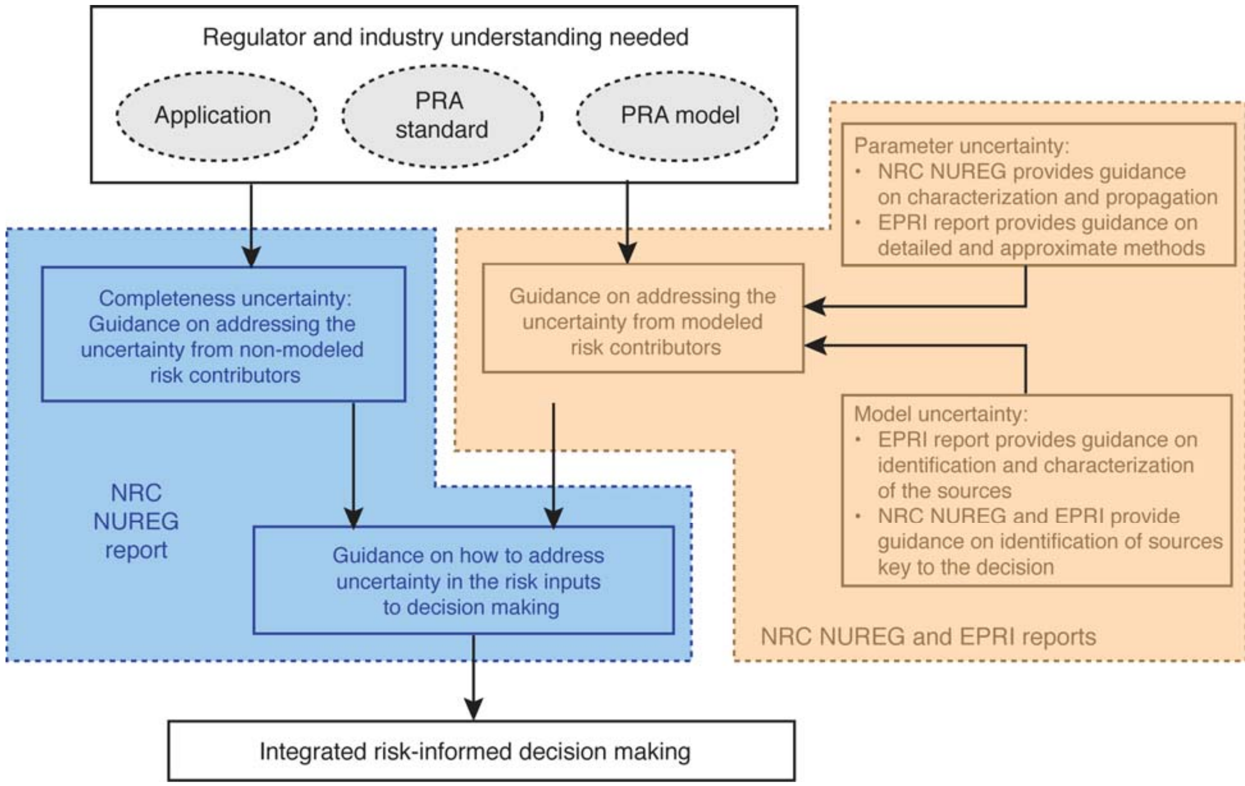

Figure 1-1
Relationship Between EPRI 1016737 and NUREG-1855

1.5 Report Organization

Combined with NUREG-1855, this report develops and provides an overall framework for the pragmatic treatment of uncertainties in risk-informed applications. Figure 1-1 summarizes the treatment of uncertainties within this document. The relationship to NUREG-1855 will be elucidated within the introduction to each section within this report.

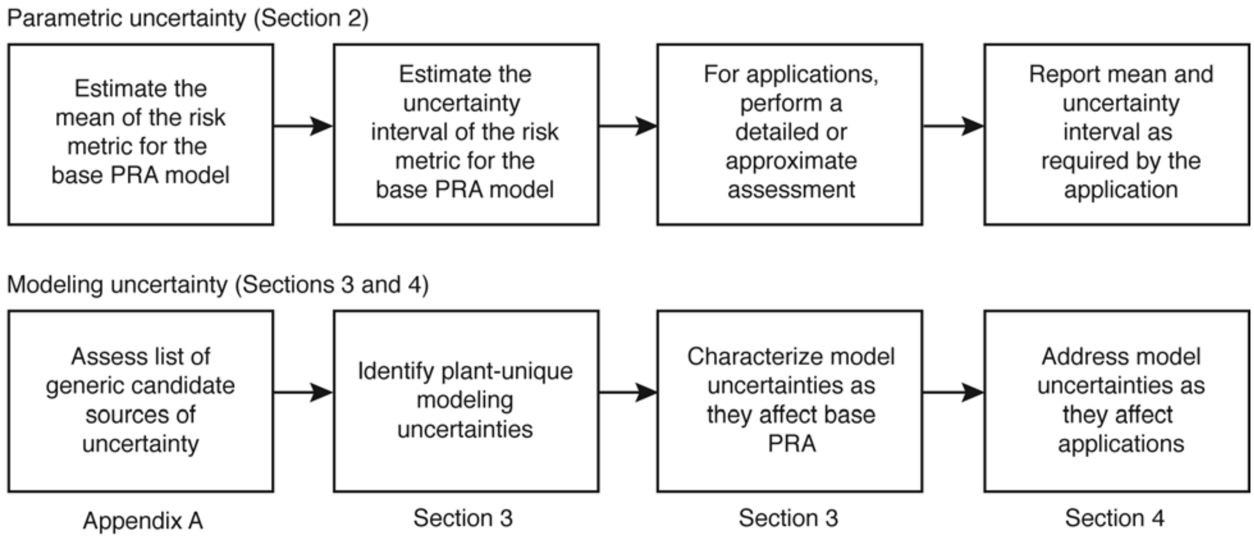

Figure 1-2
High-Level Process for Addressing Uncertainty

The report organization follows from the logical development of the basis for the assessment of parametric and modeling uncertainties.

- Section 2 provides guidance on parametric uncertainty characterization for use in meeting the ASME/ANS PRA Standard and, in particular, on addressing the state-of-knowledge correlation when evaluating the risk metrics using the PRA.
- Section 3 describes the process of how the items that originally appeared in Appendix H of the Technical Basis Document [1] were reviewed and evolved into the identification of those issues that have been earmarked as modeling uncertainty candidates based on the definitions provided above. Given that the candidate model uncertainty list exists, to meet the PRA Standard supporting requirements defined above for QU-E1, QU-E2, QU-E4, QU-F4, and LE-F3, a plant-specific issue characterization for those items in the list plus any other identified items relevant for the plant-specific base PRA model needs to be provided. This guidance also appears in Section 3 of this report.
- Section 4 provides guidance on characterizing modeling uncertainties in the context of risk-informed applications. A framework for the selection, preparation, assessment, and reporting of results of sensitivity studies to account for uncertainties in the context of decision-making is provided.
- Appendix A provides the list of those issues that were identified as sources of modeling uncertainty candidates as described in Section 3 and provides additional complementary information to meet the applicable ASME/ANS PRA Standard supporting requirements listed in Section 1.3.1 of this report.

Appendix B provides a sample implementation of the process for the base model assessment described in Section 3.

2
GUIDANCE ON TREATMENT OF PARAMETRIC UNCERTAINTIES

This section provides a summary of the information presented in the Technical Basis Document [1] and summary-level guidance that could be supported by these bases for the treatment of parametric uncertainties in nuclear power plant PRAs. Note that additional information and guidance regarding the characterization and propagation of parametric uncertainties is available in Section 4 of NUREG-1855.

2.1 Problem Statement

Many PRA calculations, including importance measures calculations, are based on the use of point estimates. The actual process is the PRA point estimate parameters from the PRA database are used to quantify the basic event probabilities and initiating event frequencies. These basic event probabilities and initiating event frequencies are the substituted into the cutset solution that is the quantified to produce the spectrum of PRA results including CDF, LERF and importance measures.

In general, the point estimates used for the input parameters should correspond to the mean values of the probability distributions representing the parametric uncertainty in those parameter values. That is, the PRA database mean values should be the result of a distribution developed from the generic or plant specific experiential component failure data or initiating event experience. If all the events in the cutset were statistically independent, (i.e., based on independent data that is not pooled or correlated in any way), the output point estimates would themselves be mean values. However, in general, this is not the case and the data used within the cutset for like components has some common element, is pooled, or is correlated in some way, and therefore the point estimate of the resulting cutset differs from the mean value due to that "correlation." This statistical phenomena is known as the state-of-knowledge correlation (SOKC) or the epistemic correlation and this phenomena causes the propagated mean value to be higher than the point estimate calculated as discussed above.

Specifically, the state-of-knowledge correlated data effect is a statistical effect that occurs when the same information is used to generate the estimates of the parameters used to evaluate the probabilities of a group of basic events (e.g., the failure to start of a group of motor driven pumps). The correlated data effect implies that when using a Monte Carlo (or similar) approach to propagate uncertainty, the same sample value drawn from the probability distribution of the parameter is used to calculate the basic event probability of all basic events within the group.

The effects of this correlation are that the true propagated mean will have a higher mean value than the point estimate value, and the parametric uncertainty about the mean will also be underestimated. Therefore, there is a desire to understand the significance of this correlation and account for it appropriately.

The characterization of parametric uncertainties is considered to be generally straightforward and within the capability of the PRA software being currently used (with some exceptions, for example, parametric uncertainty propagation of importance measures). The risk importance measures typically used and generated by current generation software packages are based on these point estimates not mean values. EPRI has previously investigated this impact on importance measures in the EPRI report, *Parametric Uncertainty Impacts on Option 2 Safety Significant Categorization* (1008905), for application to Option 2 [9].

Two supporting requirements of the ASME/ANS PRA Standard specifically address the treatment of the state-of-knowledge correlation. These two supporting requirements, QU-A3 and QU-E3 are shown in Table 2-1. Two additional SRs (LE-E4 and LE-F3) ensure that the same requirements exist for LERF as they do for CDF as also shown in Table 2-1 below.

Table 2-1
Pertinent ASME/ANS PRA Standard Supporting Requirements Related to Parametric Uncertainty

	Capability Category II	Capability Category III
QU-A3	ESTIMATE the mean CDF accounting for the “state-of-knowledge” correlation between event probabilities when significant.	CALCULATE the mean CDF from internal events by propagating the uncertainty distributions, ensuring that the "state-of-knowledge" correlation between event probabilities is taken into account.
QU-E3	ESTIMATE the uncertainty interval of the CDF results. ESTIMATE the uncertainty intervals associated with parameter uncertainties (DA-D3, HR-D6, HR-G8, IE-C15) taking into account the “state-of-knowledge” correlation.	PROPAGATE parameter uncertainties (DA-D3, HR-D6, HR-G8, IE-C15), 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.
LE-E4	QUANTIFY LERF consistent with the applicable requirements of Tables 2.2.7-2(a), 2.2.7-2(b), and 2.2.7-2(c).	
LE-F3	IDENTIFY and CHARACTERIZE the LERF sources of model uncertainty and related assumptions, consistent with the requirements of Tables 2.2.7-2(d) and 2.2.7-2(e).	

As can be seen, the Supporting Requirements direct that the mean and the parametric uncertainty interval account for the state-of-knowledge correlation effect for both CDF and LERF.

2.2 Correlated Versus Uncorrelated Effects

The correlation effect is a statistical effect that occurs when the same information is used to characterize the uncertainty distribution for a failure mode of all components of a certain type (for example, all pumps failing to run). Correlation implies that the same distribution applies to this failure mode for all of these components when they are sampled using a Monte Carlo (or similar) approach.

For example, the generic knowledge of the failure rate of a failure mode of one particular pump (such as an LPCI pump) is typically based on experience with all "similar" pumps. Therefore, the various basic events that involve this failure mode of a pump are all in fact being estimated from a single state-of-knowledge distribution. Therefore, for a completely correlated representation in a Monte Carlo (or similar) trial, this distribution should be sampled once to obtain a failure rate, and that same failure rate should be used to generate the sample value for all the pump-failure basic events in the cutset equations.

The correlation effect has some general tendencies that can be identified when it is included in the calculation of CDF. These general tendencies include the following:

- The point estimate obtained by quantification of the cutset probabilities using mean values for each basic event probability does not produce a true mean of the CDF. The Monte Carlo (or similar) simulation including the correlation effect calculates a higher mean value.
- The impact of the correlation effect on the CDF can be large when the dominant cutsets contain correlated elements.
- The larger the range factor (or width of the probability distribution) of the distributions that are correlated, the larger the effect given the same cutsets.
- The impact of the correlation effect tends to be small if there is a large number of cutsets and the correlated elements are not present in most cutsets, and the cutsets in which they occur are not dominant.
- The correlation effect increases the range factor of the resulting distribution.
- As unique plant-specific data are developed for different component applications, such as LPCI pumps versus service water pumps, the impact of the correlation effect will decrease.

Appendix F of the Technical Basis Document [1] provides the empirical calculations of the state-of-knowledge correlation and key parameters to determine if the point estimate results of the PRA are adequate to characterize the mean value. The Technical Basis Document also provides the basis to determine when the use of the point estimate result from a logic model is adequate to represent the mean of the risk metric.

An example from Appendix F of the Technical Basis Document with two correlated variables "ANDed" together in cutsets (i.e., where two correlated components are required to fail to lead directly to the risk metric) is shown in Figure 2-1. Some general insights regarding the requirements for generating less than a representatively selected 10% change in the risk metrics are summarized below.

- For correlated variables with range factor (RF – shown as error factor, EF, for the lognormal distributions utilized in Figure 2-1) less than or equal to 3, the correlation effect produces less than a 10% impact on the risk metric if the fraction of the risk metric that results from cutsets that involve correlated variables (referred to here as degree of participation) is less than approximately 18%.
- For correlated variables with RF less than or equal to 5, the correlation effect produces less than a 10% impact on the risk metric if the degree of participation in the risk metric is less than approximately 8%.
- For correlated variables with RF less than or equal to 10, the correlation effect produces less than a 10% impact on the risk metric if the degree of participation in the risk metric of the correlated variables is restricted to less than approximately 2% of the total risk metric.
- For correlated variables with RF greater than 10, the degree of participation in the risk metric is further restricted. If the range factor is 30, the degree of participation of the correlated variables in the risk metric must be much less than 1% to keep the mean within 10% of the point estimate.

Another example from Appendix F of the Technical Basis Document with three correlated variables "ANDed" together in cutsets (i.e., where three correlated components are required to fail to lead directly to the risk metric) is shown in Figure 2-2 with some general insights regarding the degree of participation requirements for a representative 10% change also summarized below:

- For correlated variables with RF less than or equal to 3, the correlation effect produces less than a 10% impact on the risk metric if the degree of participation in the risk metric is less than approximately 3%.
- For correlated variables with RF less than or equal to 5, the correlation effect produces less than a 10% impact on the risk metric if the degree of participation in the risk metric of the correlated variables is restricted to less than approximately 1% of the total risk metric.
- For correlated variables with RF greater than 5, the degree of participation in the risk metric is further restricted. If the RF is 10, the degree of participation of the correlated variables in the risk metric must be limited to much less than 1% to keep the mean within 10% of the point estimate.

Although theoretically interesting, large scale implementation of the empirical correlations provided in the Technical Basis Document can be difficult to achieve, however, and more practical guidance is desired. The intent of the next two sections is to provide such guidance.

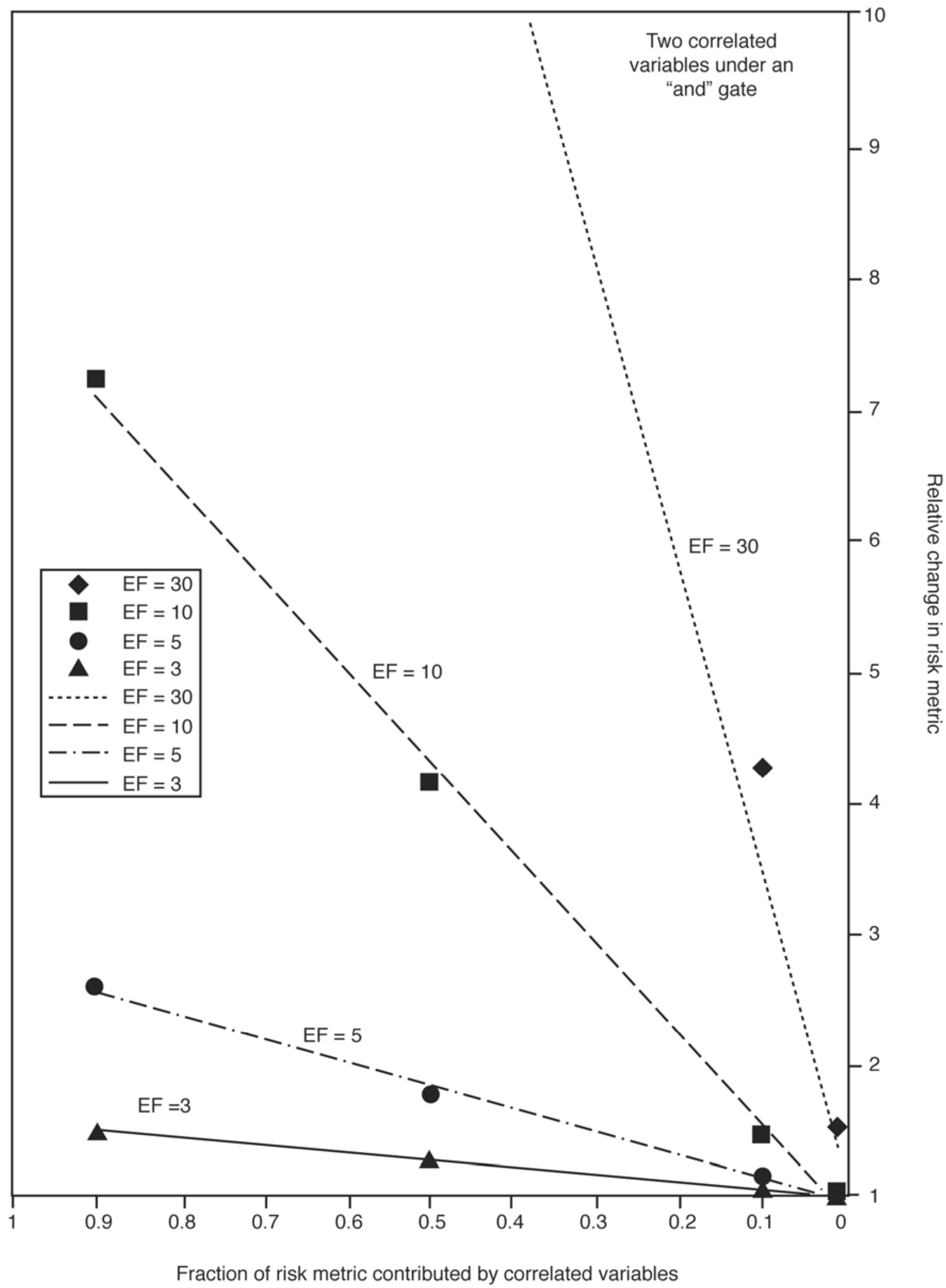

Figure 2-1
Impact of Two Correlated Variables

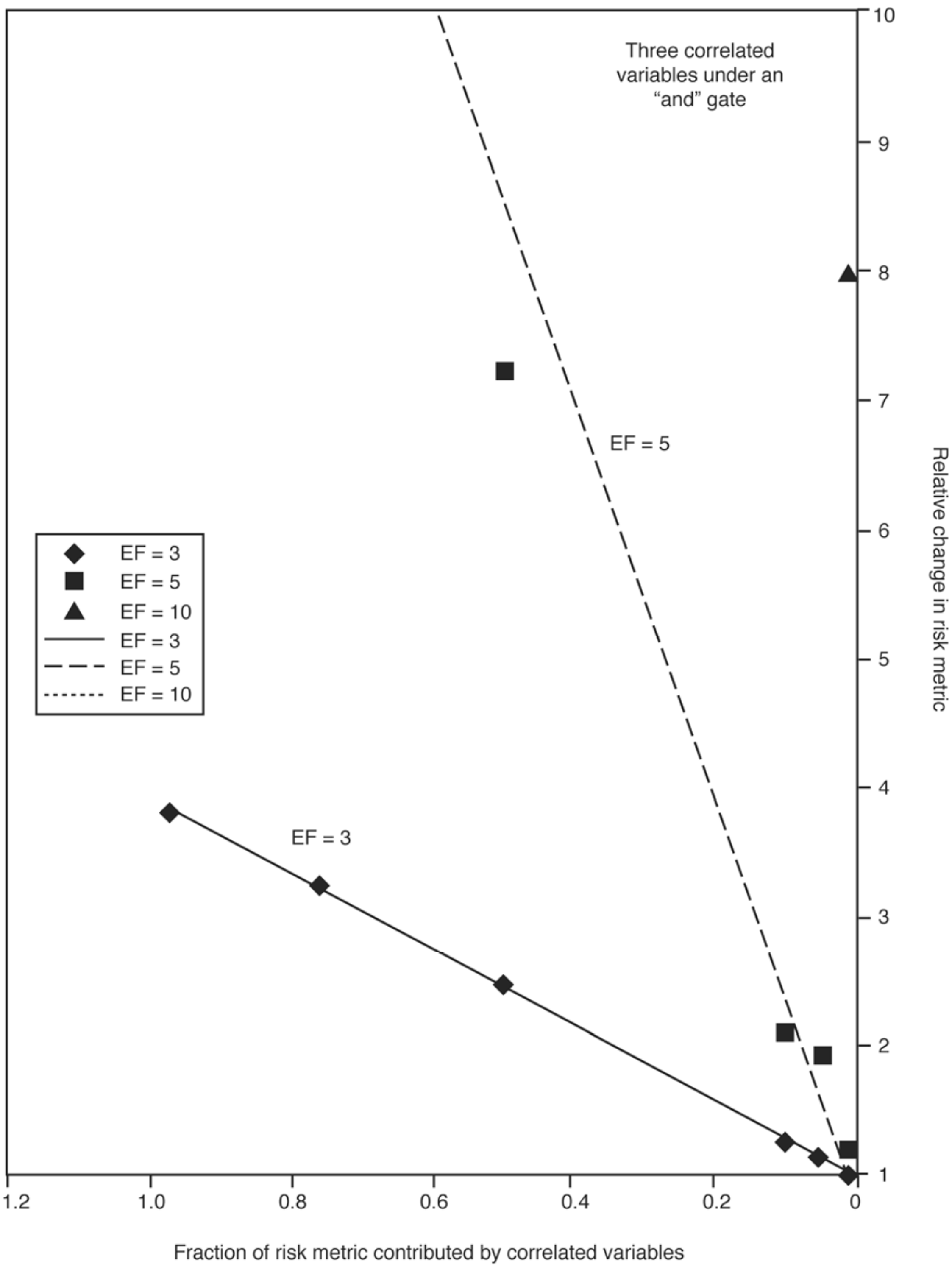

Figure 2-2
Impact of Three Correlated Variables

2.3 Development of Practical Guidance on Treatment of the State-of-Knowledge Correlation

The performance of parametric characterization includes a choice of the probability distribution and the potential state-of-knowledge related issues with the use of generic (or plant-specific) data for all similar components.

The ASME/ANS PRA Standard requires that the state-of-knowledge correlation be addressed in characterizing the mean (QU-A3) and uncertainty interval (QU-E3) for CDF and LERF, as applicable. While many computer codes are capable of handling this quantitatively (which would enable Capability Category III to be met for the QU-A3 and QU-E3 supporting requirements), not all models have been developed in a manner that allows the uncertainty propagation to be done in such a way that the state-of-knowledge correlation is accounted for correctly. Furthermore, Capability Category II of the standard does not require quantification; it simply requires "estimation" of the uncertainty interval. On the other hand, no guidance exists on how to perform this estimation in lieu of performing the full uncertainty propagation.

In general, the impact of the state-of-knowledge correlation on the baseline PRA has been seen to be fairly small (<10% increase in CDF) in several sample PRA models surveyed.

Additionally, it has been observed that the following considerations contribute to reducing the state-of-knowledge correlation effect on the CDF and LERF values:

- The ASME/ANS PRA Standard specifies the use of plant-specific data for equipment that is safety significant. The use of additional plant-specific data to generate parameter distributions for specific subsets of component types (e.g., MOVs in different systems) increases the number of sets of correlated basic events. However, the number of events for which a state-of-knowledge correlation is to be considered within the original set decreases. Therefore, the continued improvement of the supporting data used in PRA models tends to reduce the impact of the state-of-knowledge correlation effect.
- Common-cause treatment of the failure modes for multiple similar components within a correlated component type group in many cases mirrors the state-of-knowledge correlation effect and tends to be a higher contributor to the mean risk metric than the correction to the probability from the state-of-knowledge correlation.

However, it is also noted that for specific subsets of cutsets, such as those containing coincident valve ruptures for interfacing system loss-of-coolant accident (ISLOCA) evaluations, the state-of-knowledge correlation effect can be significant. Given all of these considerations, guidance for the PRA practitioner is developed for identifying when point estimate solutions are not suitable when required for mean value comparisons as described in Section 2.4.

2.4 Guidelines for Addressing Parametric Uncertainty

Figure 2-3 summarizes the high-level process for addressing parametric uncertainties. This process is applied to: 1) the estimation of the risk metric mean value for the total risk metric and individual sequences (i.e., per supporting requirement QU-A3), and 2) the estimation of the risk metric uncertainty interval (i.e., per supporting requirement QU-E3).

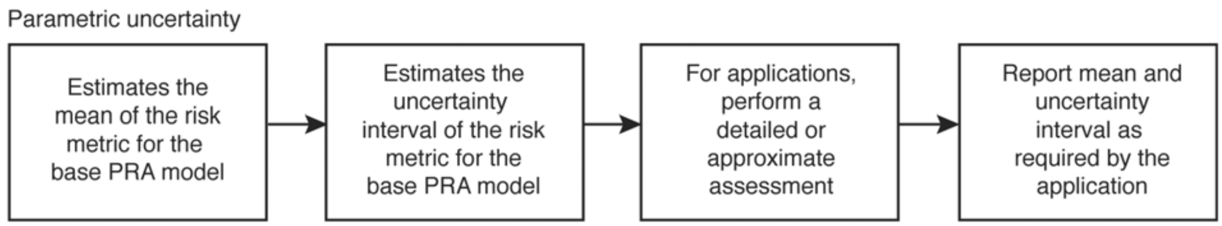

Figure 2-3
High-Level Process for Addressing Parametric Uncertainty

Two aspects of the treatment of parametric uncertainty are considered here. First, is the process of the treatment of parametric uncertainty with respect to the base PRA model and meeting QU-A3 and QU-E3 supporting requirements. Second, is the process applied to the treatment of parametric uncertainty with respect to applications. Each of these is explicitly considered in the guidelines which follow. Note, however, that not all applications require that these issues be addressed (e.g., some risk ranking application guidelines have been constructed with the general intent that the use of importance measures based on point estimates is acceptable). An example of such an application is the risk categorization guidelines in NEI 00-04 [18] developed in support of 10 CFR 50.69. In such cases, the application guidelines listed below would not need to be explicitly followed.

2.4.1 Risk Metric Mean Value Characterization

To help meet the QU-A3 supporting requirement for CDF and LERF, the following guidelines are established.

> **Guideline 1a (Base Model):**
> Ensure that the state of knowledge correlation is appropriately represented for all relevant events and perform a detailed Monte Carlo (or similar) calculation with enough samples to demonstrate convergence to calculate the mean and for information purposes, document the percentage difference from the point estimate generated by substitution of mean basic event probabilities for each basic event. This would suffice to meet Capability Category III for the QU-A3 supporting requirement and is the recommended approach to take in setting up the base PRA model.

or

Guideline 1b (Base Model):
Although not the preferred approach, if Guideline 1a cannot be completed, then perform a comparison of the plant features and model with a PRA that has previously evaluated the difference between the mean risk metric and the point estimate generated by substitution of mean basic event probabilities for each basic event, and if the plant differences and model differences are relatively minor (e.g., known differences are not significant contributors), then use the published difference from the other plant to provide a reasonable estimate of the mean risk metric taking into account the state-of-knowledge correlation. This may suffice to meet Capability Category II for the QU-A3 supporting requirement depending on the level of detail provided, but due to the potential subjectivity and complexity of such an effort compared to that required in Guideline 1a it is not the recommended approach to take in setting up the base PRA model.

When moving forward into applications, the significance of the state-of-knowledge correlation is dependent upon how the results of the PRA are being used. For applications that depend on the total CDF or LERF, then it can be expected that unless the changes to the base PRA results are focused on those cutsets involving the state of knowledge correlation, the guidelines 1a and 1b are still applicable. However, when the results required are estimates of ΔCDF or ΔLERF, the effect can be much greater depending on the subset of cutsets that is driving the results. Note that Guideline 2b may not be practical to implement in all cases since it would require a detailed review of cutsets that impact the application risk metrics to determine if state-of-knowledge correlation basic events are present, and as such, Guideline 2a represents the preferred approach in those cases.

Guideline 2a (Applications):
Similar to Guideline 1a, ensure that the state of knowledge correlation is appropriately represented for all relevant events and perform a detailed Monte Carlo (or similar) calculation with enough samples to demonstrate convergence to calculate the mean.

or

Guideline 2b (Applications):
If the risk metric used for the application is determined by cutsets that do not involve basic events with state-of-knowledge correlations in the development of the PRA (i.e. all events within the same cutset for the dominant contributors do not involve a state-of-knowledge correlation), then use the point estimate directly.

2.4.2 Risk Metric Uncertainty Interval Characterization

To help meet the QU-E3 supporting requirement for CDF and LERF, the following guidelines are established:

Guideline 3a (Base Model):
The preferred method is to perform parametric uncertainty propagation on the base PRA model using a Monte Carlo process or similar through the cutsets accounting for the state-of-knowledge correlation and report the results to establish the uncertainty bounds of 5% and 95% on the risk metric. This would suffice to meet Capability Category III for the QU-E3 supporting requirement and is the recommended approach to take in setting up the base PRA model.

or

Guideline 3b (Base Model):
Although not the preferred approach, if Guideline 3a cannot be completed, then perform a comparison of the plant features and model with a PRA that has previously evaluated the parametric uncertainty taking into account the state-of-knowledge correlation; and if the plant differences and model differences are relatively minor (e.g., known differences are not significant contributors), then report the published result from the other plant as adequately describing the parametric uncertainty about the risk metric. This may suffice to meet Capability Category II for QU-E3 depending on the level of detail provided, but due to the potential subjectivity and complexity of such an effort compared to that required in Guideline 3a it is not the recommended approach to take in setting up the base PRA model.

Having met QU-E3 to some Capability Category, the uncertainty interval on the base PRA model is established. However, most applications of the PRA model do not require that the uncertainty interval be provided to the decision maker. If it is required within the guidelines for that application, then the following guidance is provided in the context of applications. Note, however, that Guideline 4a may not be practical to prove in all cases, and as such, Guideline 4b represents the preferred approach when the uncertainty interval is a desired input by the decision maker.

Guideline 4a (Applications):
Demonstrate that the range of the uncertainty interval is not expected to significantly change (e.g., because the significant contributors for the application do not involve state-of-knowledge correlated basic events) from the base model uncertainty interval and report the base model uncertainty interval.

or

Guideline 4b (Applications):
If the conditions in Guideline 4a are not met, then perform a parametric uncertainty propagation using a Monte Carlo process or similar through the cutsets accounting for the state-of-knowledge correlation and report the results to establish the uncertainty bounds of 5% and 95% on the risk metric.

3
GUIDANCE FOR THE IDENTIFICATION OF SOURCES OF MODELING UNCERTAINTY FOR THE BASE MODEL ASSESSMENT

This section of the report is intended to be consistent with and complementary to Section 5.2 of NUREG-1855. The major objective of the guidance in this section is to provide the necessary details to identify and characterize the sources of model uncertainty for the base PRA model. The results of this initial assessment are then utilized as one input to the process for identifying potentially key sources of uncertainty in applications as described below.

3.1 Framework

There are multiple motivations for identifying the model inputs that are key contributors to uncertainty in model outputs. First, an identification of significant contributors to the output risk metrics gives the analyst an awareness of which inputs drive the results. The nature of PRA is such that the answer has inherent uncertainty that must be appreciated in using the results and making decisions. Second, a basic exploration of the models, inputs, and results promotes improved understanding and interpretation of the analysis for the decision maker.

Modeling uncertainties come about because of limitations in our knowledge of issues that are represented in a comprehensive analysis, and they can manifest themselves in the methods and assumptions used in developing the PRA. In most probabilistic assessments, the majority of the uncertainty in the output is attributable to uncertainty in a small subset of the inputs. An identification of this subset of highly significant contributors to output uncertainty has useful practical implications. It enables the application of resources for the characterization of uncertainty on a small number of important inputs, rather than spread resources thinly across the entire set of inputs entering the analysis. This observation supports the use of a focused effort on identification and characterization of sources of modeling uncertainty. The trade-off in resource expenditure for the characterization of uncertainties would appear to be prudently made through a process that screens out those contributors that have relatively small influences on the risk metrics and associated decisions. This allows the focusing of resources on the evaluation and characterization of the sources of modeling uncertainty to meet the related supporting requirements from the PRA Standard and to be prepared for identifying potential key sources of uncertainty in applications.

A framework is established for the development of guidance on the treatment of modeling uncertainties for the purposes of supporting a risk-informed decision. Modeling decisions are considered to be those outside of the decisions already well characterized by parametric uncertainty. The purpose is to provide guidance on: 1) identifying applicable modeling uncertainties and 2) the approaches and methods for dealing with modeling uncertainties.

The framework is formulated to be consistent with the requirements and guidance in the ASME/ANS PRA Standard, RG 1.174, and RG 1.200. For example, the modeling uncertainty evaluation makes use of both qualitative and quantitative evaluations and the use of sensitivity studies. All of these are consistent with the ASME/ANS PRA Standard, RG 1.200, and RG 1.174. It is recognized, however, that this approach does not constitute a complete quantitative uncertainty analysis with development of all alternative hypotheses and weighting factors applied to each hypotheses using some expert elicitation or other method for all of the modeling uncertainties. Such an approach is not deemed practical to fully implement.

Figure 3-1 provides a generalized framework for identifying, characterizing, and addressing the modeling uncertainties.

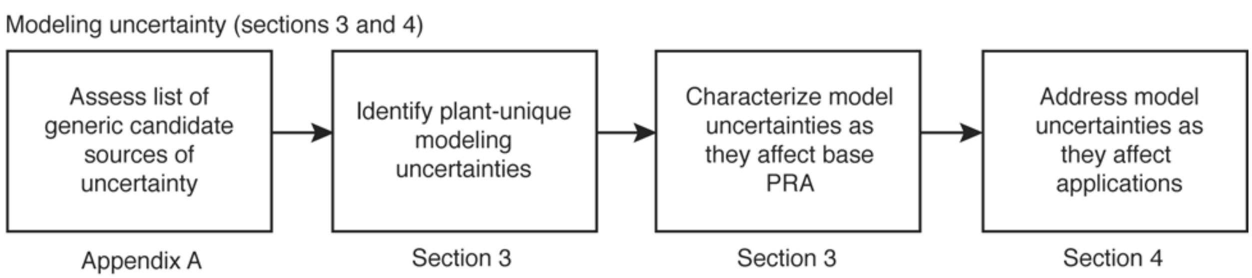

Figure 3-1
High-Level Process for Addressing Modeling Uncertainty

3.1.1 Base Probabilistic Risk Assessment Model

The motivation in performing this assessment in the base PRA model is twofold. First, it provides a mechanism for fully meeting the ASME/ANS PRA Standard supporting requirements delineated in Section 1.3.1 of this report. More importantly, though, it provides the PRA analyst with a more thorough understanding of the model that will ultimately be used in risk-informed applications.

The framework of the approach for treatment of modeling uncertainty has been developed and implemented from the high-level concept shown in Figure 3-1 for the base PRA model is to make use of the following structured process as shown in Figure 3-2.

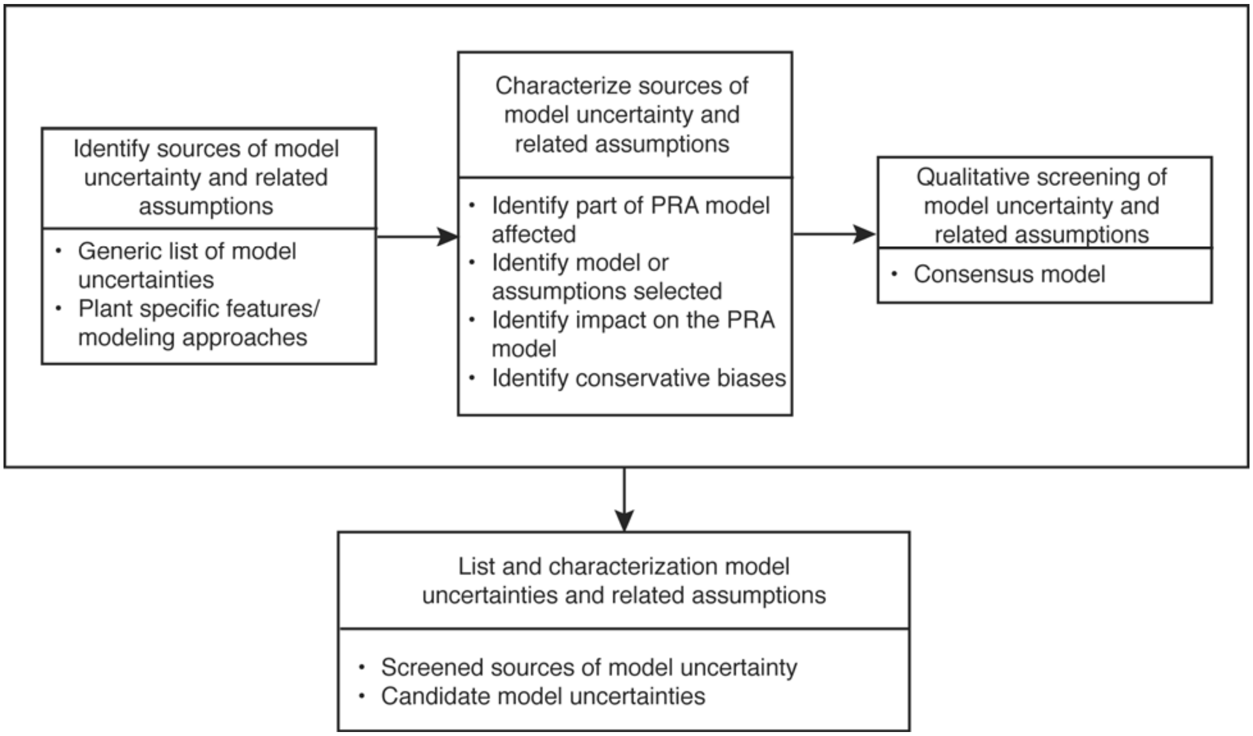

Figure 3-2
Model Uncertainty Identification, Characterization, and Screening for Base Probabilistic Risk Assessment Models

Identify of Sources of Modeling Uncertainty and Related Assumptions (Identification)

1. Identify generic contributors to modeling uncertainty using the ASME/ANS PRA Standard as a structure. These generic candidates include those issues that have been earmarked as modeling uncertainty candidates based on the definitions provided in Appendix A. The objective is to identify those sources of uncertainty with the highest potential to significantly change the risk metric.

2. Evaluate the applicability of generic model uncertainties to the specific plant and PRA to provide the final generic list to be reviewed as part of the plant-specific determination of modeling uncertainties.

3. Examine plant-specific features/modeling approaches for additional uncertainties to identify if there are plant-specific treatments or PRA modeling that introduce uncertainties not included on the generic list. Add any plant specific sources of uncertainty or related assumptions to develop the plant-specific list.

Characterize Sources of Model Uncertainty and Related Assumptions (Characterization)

4. The part of the PRA model that is affected by the source of model uncertainty or related assumption needs to be identified. This characterization is necessary since not every part of the PRA is involved in every application of the model. The part of the PRA model affected can be the basic event level, in specific portions of the system logic structure, or in specific portions of the accident sequence modeling.

5. The list of related assumptions or models are identified to properly characterize how the source of uncertainty is represented in the PRA model.
6. The impact on the PRA model provides a characterization of how the related assumptions or chosen models will affect the PRA model basic event values, system logic structure, or accident sequence modeling.
7. Identify conservative biases. This step provides a method to characterize the candidate modeling uncertainties so that they are less likely to become a key modeling uncertainty in applications. It is critical at this stage to ensure that the conservative bias in a particular candidate model does not unduly influence the overall PRA model.

   Conservatism may be a means to addressing uncertainties in some cases, but it is not a panacea. Excessive conservatism is an anathema to a realistic assessment of risk. Conservatism is often used as a means to address uncertainties in minor contributors. Examples include the following:

   - No credit for recovery from loss of support system initiating events
   - No credit for operation of equipment without dc that normally requires dc for operation
   - Unmitigated ATWS scenarios with LOCA conditions from the initiator or from consequential overpressure failures are assigned directly to core damage
   - Room cooling always assumed to be required even though there may be some conditions where it would not be required

   Judiciously applied realistic conservatism has the potential to provide a PRA that avoids many of the traps associated with the use of excess conservatism. However, systematically introducing conservatism can create a bias. In turn, biases skew results (e.g., in component importance measures or relative accident sequence contributions). This can be misleading and mask important risk contributors.

Qualitative Screening of the Sources of Uncertainty and Related Assumptions (Screening)

8. Apply consensus model. This step makes use of those areas of the PRA where extensive historical precedence is available to establish a model that has been accepted and yields PRA results that are considered reasonable and realistic.

   This is consistent with the definition of model uncertainties and consensus models provided in NUREG-1855.

   > A *source of model uncertainty* is one that is related to an issue in which there is no consensus approach or model and where the choice of approach or model is known to have an effect on the PRA model (e.g., introduction of a new basic event, changes to basic event probabilities, changes in success criterion, introduction of a new initiating event).
   >
   > [A] *consensus model,* in the most general sense, [is] a model that has a publicly available published basis and has been peer reviewed and widely adopted by an appropriate stakeholder group. In addition, widely accepted PRA practices may be regarded as consensus models. Examples of the latter include the use of the constant probability of

failure on demand model for standby components and the Poisson model for initiating events. For risk-informed regulatory decisions, the consensus model approach is one that the NRC has utilized or accepted for the specific risk-informed application for which it is proposed.

Base Model Candidate Model Uncertainties

9. Those candidate model uncertainties that are not dispositioned using consensus models (Step 8) are carried forward as applicable candidate model uncertainties from the base PRA model assessment. Note that for those items identified as representing conservative biases (Step 7), if removing the identified conservative bias would make an already acceptable calculated risk metric more acceptable compared to the acceptance guidelines (e.g., ΔCDF), then that source of uncertainty can generally be screened from being a potential key source of uncertainty in that application of the model.

3.2 Identification of Candidate Sources of Uncertainty on a Generic Basis

A generic list of potential model uncertainties is developed through the examination of the ASME/ANS PRA Standard and available industry/NRC PRAs (see Appendix H of the Technical Basis Document). The derivation of the list of candidate uncertainties makes use of two complementary approaches to ensure that the generic list is robust. Figure 3-3 indicates that the identification process examines the following to provide a structure and cross checking of the potential candidates for model uncertainty consideration: 1) root causes of uncertainty and 2) ASME/ANS PRA Standard high-level requirements.

Upon development of the initial broad list it became apparent that most of the items are related to scope or level of detail issues that can relate to simplifying assumptions rather than true modeling uncertainty issues. An assumption related to scope or level of detail is simply one that is made for modeling convenience and results in a simplification of the analysis performed. Alternatively, true modeling uncertainties lead to assumptions that are made with the knowledge that a different reasonable alternative assumption exists. A *reasonable alternative* assumption is one that has broad acceptance within the technical community and for which the technical basis for consideration is at least as sound as that of the assumption being made.

The characteristics associated with this definition of model uncertainty include the following:

- The phenomena or nature of the event or failure mode is not completely understood
- Significant interpretations to infer behavior are required to develop a model (this is the case where some information is available, but is not sufficient to derive a definitive model or value)
- There is a general agreement that the issue represents a potential source of modeling uncertainty

The results of applying the process shown in Figure 3-4, the list of candidate generic sources of uncertainty as well as those earmarked as candidate sources of *model uncertainties*, are an input to the modeling uncertainty assessment process described here. Additionally, for each of the *model uncertainty* issues an indication is provided for: (1) typical parts of the model affected, (2) representative sample approaches, and (3) the resulting impact on the model in Appendix A.

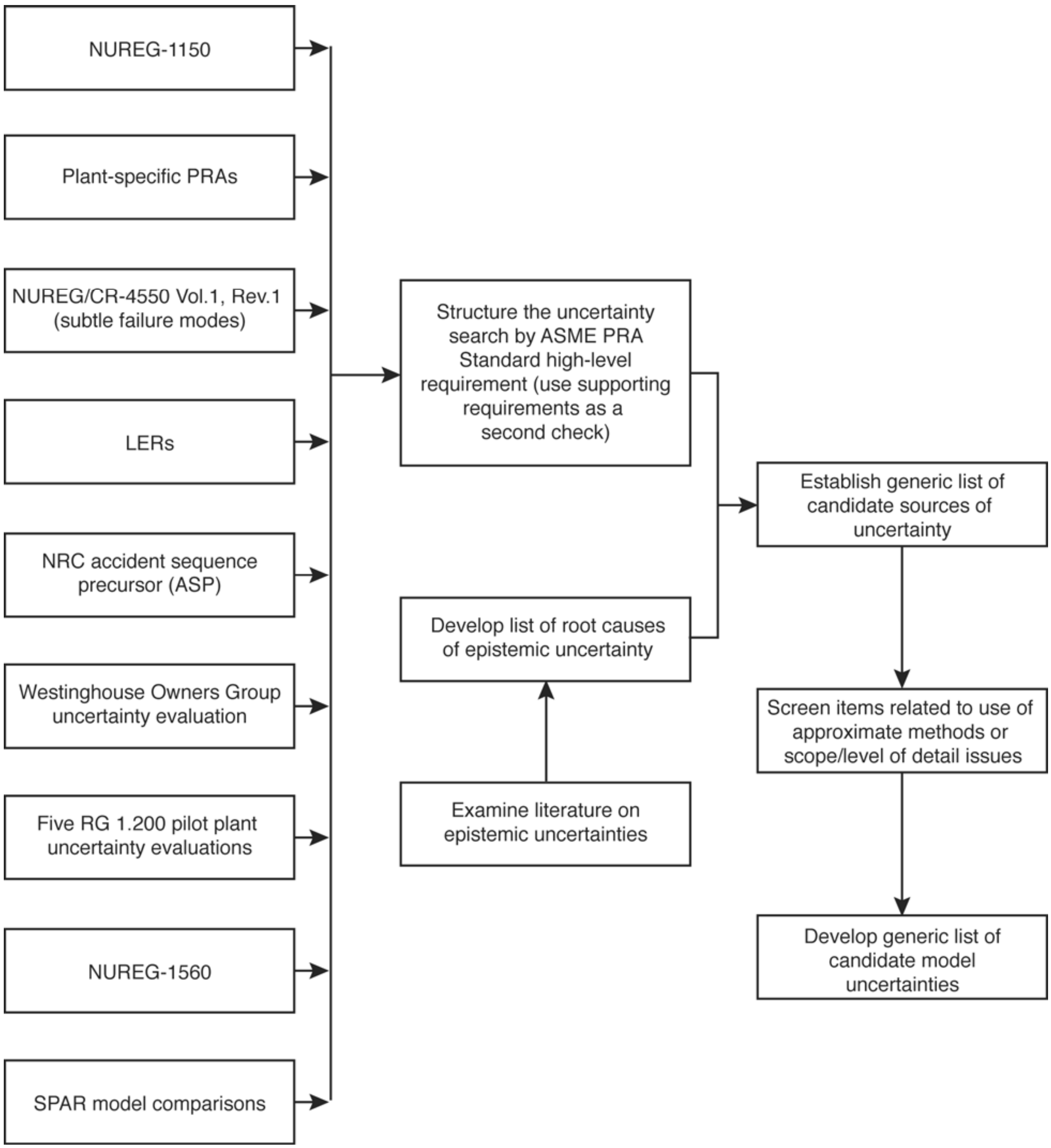

Figure 3-3
Development of Candidate List of Modeling Uncertainties

3.3 Identification of Plant-Specific Modeling Uncertainties

An examination of plant-specific features and modeling approaches is required to ascertain if there are plant-specific features or modeling approaches that introduce uncertainties not incorporated into the generic list. This portion of the assessment allows for the identification of any plant-specific features that require unique phenomenological assessments not considered in the generic list in Appendix A.

Some examples of plant-specific features that may introduce unique modeling uncertainty issues are listed below:

Plant-Specific Feature	Source of Modeling Uncertainty
• Cavity sump drain discharge directly to auxiliary building	• How one accounts for core debris to bypass containment and enter auxiliary building given accident sequence conditions that allow core debris to collect in the cavity sump
• Steam line location relatively close to the RPV high level trip set point and turbine driven system susceptibility to water ingress	• How to account for conditions that allow for water in the steam line and the impact on the connected turbine driven systems
• Reliance on containment overpressure to preserve ECCS NPSH requirements	• How to account for conditions that would potentially eliminate the availability of containment overpressure
• Utilization of modeling approaches that differ from available consensus model approaches	• How to account for the development and progression related to the use of a consensus approach versus an alternate approach

When these items are identified they are added to the list of model uncertainty items included in Appendix A and addressed in a similar fashion as outlined below. At a minimum, the plant-specific model uncertainty identification should include any items that were determined using the requirements in paragraph 1.4.3 of the ASME/ANS PRA Standard for implementing an expert judgment process.

3.4 Assessment of Sources of Modeling Uncertainties on a Plant-Specific Basis

It is recognized that there are modeling uncertainties associated with the representation of the complex physical systems, structures, and components of a nuclear power plant. This recognition has led the NRC and industry to formulate a risk-informed process that incorporates defense-in-depth considerations into the risk-informed process. The defense-in-depth process is an important technique that supplements the PRA and provides confidence that a blended approach to risk-informed regulation properly addresses uncertainties. In addition to the reliance on defense-in-depth, this report provides methods and approaches for the formal incorporation of modeling uncertainties explicitly into the PRA model construction and subsequent evaluations. In addition, the parametric uncertainty evaluation already recognizes that the probabilistic results have a distribution associated with them and that the results are subjected to differences imposed by these parametric uncertainties.

The candidate sources of modeling uncertainty have been identified based on interactions with NRC and industry personnel. The list has been developed with the intent of defining that set of issues that need to be addressed to satisfy the following supporting requirements from the ASME/ANS PRA Standard [26].

- QU-E1: IDENTIFY sources of model uncertainty.
- QU-E2: IDENTIFY assumptions made in the development of the PRA model.
- QU-E4: For each source of model uncertainty and related assumption identified in QU-E1 and QU-E2, respectively, IDENTIFY how the PRA model is affected (e.g., introduction of a new basic event, changes to basic event probabilities, change in success criterion, introduction of a new initiating event)
- QU-F4: DOCUMENT the characterization of the sources of model uncertainty and related assumptions (as identified in QU-E4).
- LE-F3: IDENTIFY and CHARACTERIZE the LERF sources of model uncertainty and related assumptions, consistent with the requirements of Tables 2.2.7-2(d) and 2.2.7-2(e).

In the process of identifying and characterizing assumptions associated with the potential sources of modeling uncertainty, it is important to establish a definition for such assumptions. These definitions have also been provided in the ASME/ANS PRA Standard [26].

> An *assumption* is a decision or judgment that is made in the development of the PRA model. An assumption is either related to a source of model uncertainty or is related to scope or level of detail.
>
> An *assumption related to a model uncertainty* is made with the knowledge that a different reasonable alternative assumption exists. *A reasonable alternative assumption* is one that has broad acceptance within the technical community and for which the technical basis for consideration is at least as sound as that of the assumption being made.
>
> An *assumption related to scope or level of detail* is one that is made for modeling convenience.

Note that the definitions above provide delineation for those sources of model uncertainty (and related assumptions) that should be the focus for meeting the QU supporting requirements described above (as opposed to assumptions related to scope or level of detail). Also note that the types of reasonable alternative assumptions related to sources of model uncertainty can lead to increases or decreases in the calculated risk metrics, and the process should recognize that both outcomes are possible. The development of the base PRA model therefore requires a careful balance of conservative bias treatment with realistic assessments to ensure that the base PRA model risk profile is adequately developed. Applications of the PRA model could become more or less dependent on the choices made for addressing modeling uncertainty. The intent is to assess on a plant-specific basis those issues that need to be identified and characterized to meet the supporting requirement definitions provided above.

Section 3.2 describes the process of how the items that originally appeared in Appendix H of the Technical Basis Document were reviewed and evolved into the identification of those issues that have been earmarked as modeling uncertainty candidates based on the definitions provided above. Given that the candidate model uncertainty list exists, to meet the PRA Standard Supporting Requirements defined above for QU-E1, QU-E2, QU-E4, QU-F4, and LE-F3 a plant-specific issue characterization needs to be provided. This could take the form shown in Figure 3-4 below.

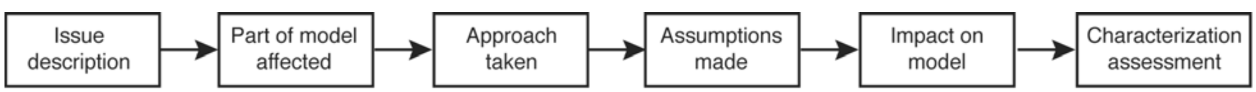

Figure 3-4
Identifying and Characterizing Model Uncertainty Issues

To assist with this process, Figure 3-5 provides an overview for the issue characterization that is provided for each model uncertainty item in Appendix A. That is, for each model uncertainty issue that has been identified, an indication is provided for: (1) a discussion of the issue, (2) typical parts of the model affected, and (3) representative sample approaches. These three aspects are included in Table A-1 of Appendix A where the impact on the model and the characterization assessment (e.g., applied conservative bias or applied consensus model), is left to be uniquely defined for each model.

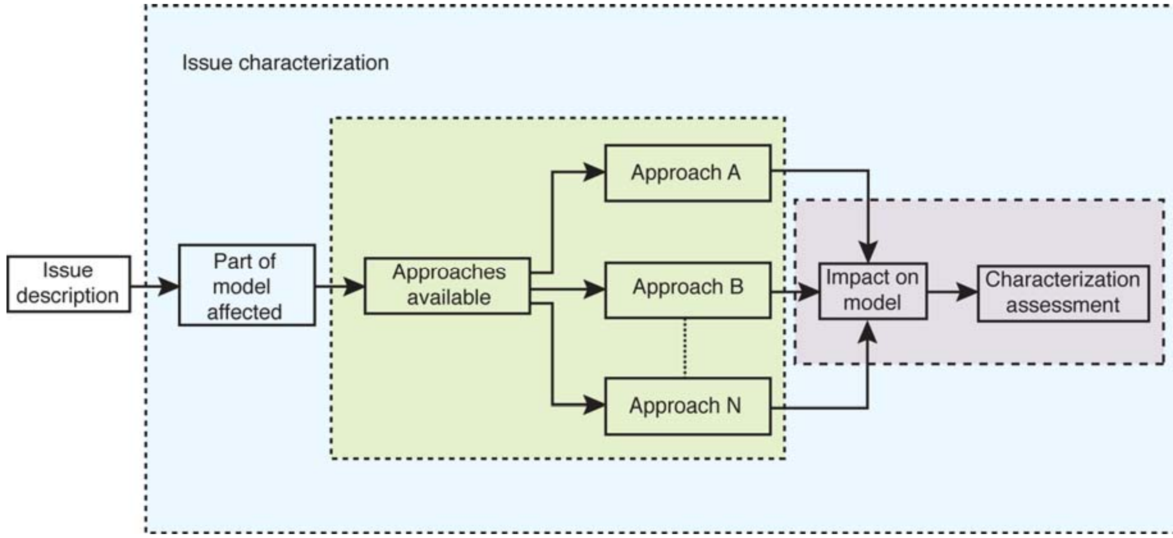

Figure 3-5
Template for Model Uncertainty Issue Characterization

The intent of Table A-1 in Appendix A is to provide a template for meeting the PRA Standard Supporting Requirements defined above for QU-E1, QU-E2, QU-E4, QU-F4, and LE-F3. That is, for each applicable model uncertainty item, a plant-specific issue characterization should be provided to fully satisfy the related supporting requirements. In general, information from one of the typical approaches described in Appendix A could be applicable to the plant-specific issue characterization development. After that, only the impact on the model and characterization assessment would need to be uniquely defined.

A sample application of the process described in Appendix A for one of the sources of model uncertainty is shown below in Section 3.4.1.

3.4.1 Example Implementation of the Process

One of the identified generic sources of model uncertainty is the topic of core cooling success following containment failure or venting through non-hard pipe vent paths. The model uncertainty issue characterization for a representative BWR Mark II plant is shown below in Table 3-1. This is one example of how the process of performing the base model assessment to identify and characterize the sources of modeling uncertainty would be employed. A complete assessment for the representative BWR Mark II plant is shown in Appendix B.

Table 3-1
Representative Issue Characterization for a Source of Model Uncertainty

Topic (QU-E1)	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made (QU-E2)	Impact on Model (QU-E4)	Characterization Assessment
Core cooling success following containment failure or venting through non-hard pipe vent paths	Loss of containment heat removal leading to long-term containment over-pressurization and failure can be a significant contributor in some PRAs. Consideration of the containment failure mode might result in additional mechanical failures of credited systems. Containment venting through “soft” ducts or containment failure can result in loss of core cooling due to environmental impacts on equipment in the reactor building, loss of NPSH on ECCS pumps, steam binding of ECCS pumps, or damage to injection piping or valves. There is no definitive reference on the proper treatment of these issues.	Long-term loss of decay heat removal scenarios	*With containment venting unsuccessful:* Limited credit is taken for continued injection immediately before and after containment failure from the CRD system only. CRD is the only viable injection source as the containment pressure rises above the vent pressure because the SRVs will close on high containment pressures and the RPV will re-pressurize above the low pressure injection capabilities.	1. Low pressure injection sources internal to containment (LPCI and Core Spray from the suppression pool) are assumed to be lost before containment failure when the RPV re-pressurizes above their discharge pressure limits and are also assumed to fail after containment failure due to the items listed in the discussion of the issue.	1. LPCI and Core Spray are not credited for success after containment failure.	No credit for these systems after containment failure may represent a slight conservative bias slant. This should not be a source of model uncertainty in most applications.
				2. HPCI and RCIC are assumed to be unavailable prior to containment failure since high pool temperatures would preclude their use and the RPV would be depressurized per procedure prior to the SRVs re-closing and then after the SRVs can re-open as the containment depressurizes following containment failure.	2. HPCI and RCIC are not credited for success after containment failure.	No credit for these systems after containment failure may represent a slight conservative bias slant. This should not be a source of model uncertainty in most applications.
				3. Other alternate low pressure injection systems are not credited for success since they cannot inject during the time frame prior to containment failure when the SRVs close and the RPV re-pressurizes above their shutoff head.	3. Other alternate low pressure injection systems (e.g., Fire Water and RHRSW through RHR) are also not credited for success after containment failure.	No credit for these systems after containment failure represents a slight conservative bias slant. This should not be a source of model uncertainty in most applications.

Table 3-1 (continued)
Representative Issue Characterization for a Source of Model Uncertainty

Topic (QU-E1)	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made (QU-E2)	Impact on Model (QU-E4)	Characterization Assessment
				4. Following containment failure, injection from CRD could still be maintained, but if a large containment failure occurs low in the reactor building, CRD is also assumed to be lost. This failure probability is based on a detailed structural analysis of the Mark II containment design.	4. CRD is credited for success after containment failure, but an additional basic event is included that represents the likelihood that the containment failure size and location disrupts the capability of CRD to inject.	CRD injection capability after containment failure is identified as a candidate source of model uncertainty.
			With containment venting successful: As with the containment failure cases, low pressure injection is required for use up the time of reaching the containment vent pressure. This can be satisfied with LPCI or Core Spray from the suppression pool. If containment venting is successful, then a myriad of low pressure external injection systems could also provide an adequate supply of inventory for RPV makeup following containment depressurization.	1. It is assumed that injection sources from the suppression pool will be lost even though they are more likely to survive since a controlled depressurization could occur in these cases compared to the containment failure cases.	1. LPCI and Core Spray are not credited for success after containment venting.	No credit for these systems after containment venting represents a slight conservative bias treatment. This should not be a source of model uncertainty in most applications.
				2. Potentially viable injection systems post-venting include CRD, Condensate, Fire Water or RHRSW through RHR.	2. Logic is included in the post-venting portion of the event sequence modeling for providing RPV injection from these systems. CRD and Condensate also require make-up to the CST or hotwell, respectively.	Slight conservative bias treatment given that no other systems are credited and fire water and RHRSW are also not credited (see #3 below). This should not be a source of model uncertainty in most applications.
				3. However, since no specific direction is included to line-up the alternate injection systems prior to venting containment, the conditions in the reactor building post-venting are assumed to preclude their use.	3. A separate basic event for lining up alternate injection from fire water or RHRSW prior to venting is included in the model that is currently set to guaranteed fail.	No credit for these systems after containment venting represents a slight conservative bias treatment. This should not be a source of model uncertainty in most applications.

Additionally, a slightly different approach is recommended to address the other model uncertainty items identified in Table A-2 where these three issues were identified as potential sources of model uncertainty since they are generally understood and accepted as areas of uncertainty that can be significant contributors to CDF and LERF. As described in Appendix A, these three additional model uncertainty issues are:

- Basis for human error probability (HEP) values
- Treatment of human failure event dependencies
- Intra-system common cause events

As part of the lessons learned in the original invocation of the process outlined in the Application Guide [2], a recommendation was made to include the development of a standard set of sensitivity cases to perform that may envelope several potential sources of uncertainty at a relatively high level. Developing the set of sensitivity cases in lieu of trying to identify and characterize all potential sources of uncertainty associated with these issues has the potential benefit of highlighting the potential impact of these specific issues prior to performing applications. Therefore, a standard set of four sensitivity cases is recommended as follows:

- All HEP probabilities (including pre-initiators, post-initiators, and dependent HEP values) set to their 5th percentile value (the use of zero-value HEP probabilities is also deemed acceptable)
- All HEP probabilities (including pre-initiators, post-initiators, and dependent HEP values) set to their 95th percentile value
- All CCF probabilities set to their 5th percentile value (the use of zero-value CCF probabilities is also deemed acceptable)
- All CCF probabilities set to their 95th percentile value

Although the 95th percentile values may not represent the absolute upper bounds for each individual HEP or CCF event (e.g., if alternate models would lead to very different results), the aggregate impact of the results of these analyses can be compared to the RG 1.174 CDF and LERF limits of $1x10^{-4}$/yr for CDF and $1x10^{-5}$/yr for LERF to obtain insights into the sensitivity of the base PRA model results to these generic high level sources of modeling uncertainty. Similarly, the lower bound (or better the zero value) sensitivity cases provide additional useful information regarding the relative contribution of each of these high level sources of uncertainty to the overall risk profile.

3.4.2 Summary of Implementation of the Process

The structure and format of Table 3-1 can be reproduced for each of the applicable items identified in Appendix A (Table A-1, as well as any other plant unique features that need to be addressed) to provide a plant-specific model uncertainty issue characterization to meet the PRA Standard Supporting Requirements defined above for QU-E1, QU-E2, QU-E4, QU-F4, and LE-F3. Additionally, the PRA model documentation would need to describe the basis and methods used for the three additional model uncertainty items in Table A-2. However, the presentation of the results of the sensitivity cases described above would provide added assurance of model understanding prior to performing applications. An example implementation of the recommended process for all of the applicable items in Table A-1 and the results from the sensitivity studies for the issues included in Table A-2 is shown in Appendix B of this report.

4
CHARACTERIZATION OF SOURCES OF UNCERTAINTY FOR RISK-INFORMED APPLICATIONS

This section of the report is intended to be consistent with and complementary to Sections 5.3 and 5.4 of NUREG-1855. The major objective of the guidance in this section is to provide the framework to identify and assess sources of uncertainty in applications. Additionally, this section includes guidance on the presentation of results to the decision maker. However, since this report does not address completeness uncertainty, the overall guidance in Sections 6 and 7 of NUREG-1855 should be followed when preparing a comprehensive risk-informed PRA application.

4.1 Definition of Key Source of Uncertainty

Consistent with RG 1.200, Revision 1 (and later), a source of uncertainty or assumption can become a key source of uncertainty or a key assumption only in the context of an application. The ASME/ANS PRA Standard provides clarification on this issue.

- A *source of model uncertainty* is one that is related to an issue in which there is no consensus approach or model and where the choice of approach or model is known to have an effect on the PRA model (e.g., introduction of a new basic event, changes to basic event probabilities, change in success criterion, introduction of a new initiating event).
- A source of model uncertainty is labeled *key* when it could impact the PRA results that are being used in a decision, and consequently, may influence the decision being made. Therefore, a key source of modeling uncertainty is identified in the context of an application. This impact would need to be significant enough that it changes the degree to which the risk acceptance guidelines are met, and therefore, could potentially influence the decision. For example, for an application for a licensing base change using the acceptance guidelines of RG 1.174, a source of model uncertainty or related assumption could be considered “key” if it results in uncertainty regarding whether the results lie in Region II or Region I, or if it results in uncertainty regarding whether the result becomes close to the region boundary or not.

Note that the definitions above indicate that key assumptions can be from assumptions related to a source of model uncertainty (as evaluated for the base model as described in Section 3) or from application-specific sources of model uncertainty. The framework for the assessment of sources of uncertainty for applications therefore needs to account for both of these possibilities. Such a process is outlined in Section 4.2.

4.2 Framework for the Assessment of Sources of Uncertainty for Applications

Risk-informed applications may present different and unique issues versus the base model that need to be discussed to ensure that the model uncertainties are appropriately treated. Many model uncertainties may not influence the decision on a specific application. There may also be new issues introduced by the application or scope or level of detail assumptions and sources of uncertainty that qualify the issue as a potential key source of uncertainty.

Figure 4-1 provides the framework for the assessment of model uncertainties for applications.

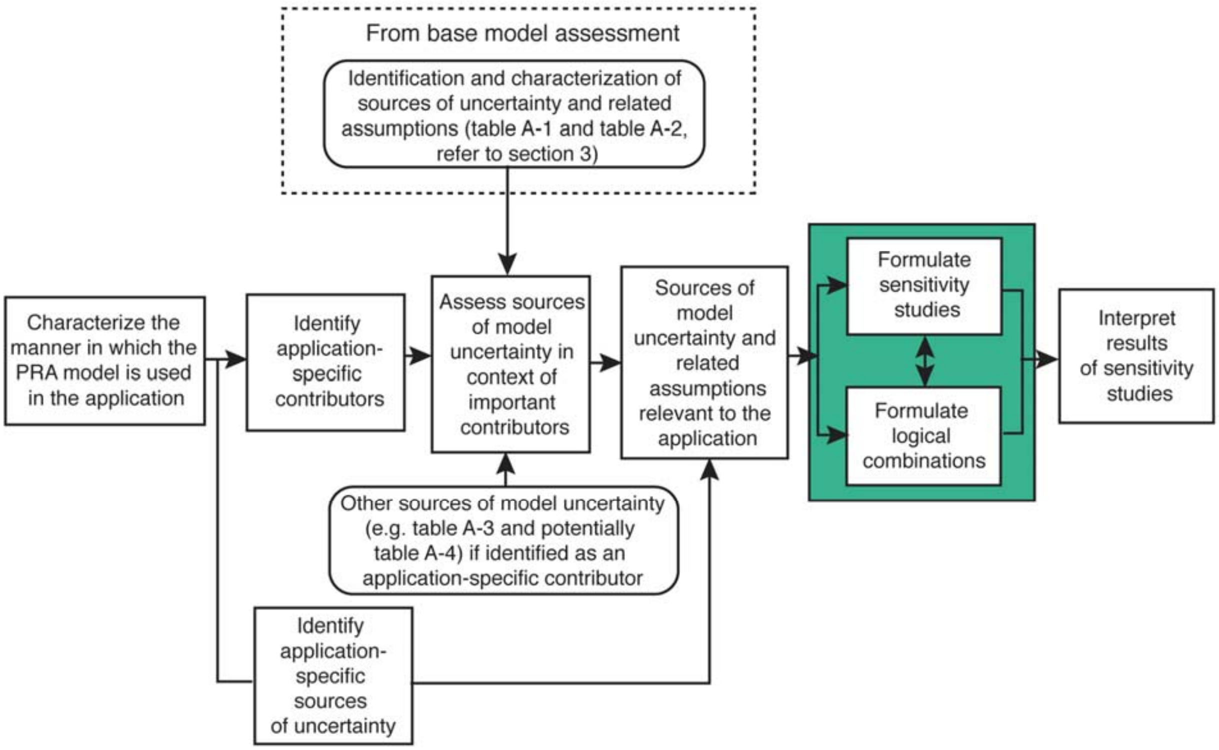

Figure 4-1
Framework for the Assessment of Model Uncertainties for Applications

The process steps include the following:

1. Characterize the manner in which the PRA model is used in the application

The modeling uncertainties that may influence the risk metrics used to assess applications may be different from those identified in the base PRA model.

Therefore, a process that is explicitly oriented towards the unique aspects of the application is desired. The characterization of the application requires:

- The qualitative description of the application is in a clear manner that allows its interpretation in the quantitative PRA model.

The risk metric being evaluated will also be a principal consideration (for example, CDF/LERF, ΔCDF/ΔLERF, and importance measures).

2. Characterize modifications to the PRA model

The quantitative PRA model for the application should be consistent with the base model. However, there may be changes that are made to the PRA model to perform the PRA model assessment for the application. These inputs can be reflected in such areas as follows:

- Changes to the level of detail incorporated into the model for the application that introduces new model uncertainty issues
- Other model changes incorporated solely for the assessment in the specific application that introduces model uncertainty issues

These other sources of uncertainty and related assumptions need to be identified and characterized and are carried forward directly as relevant to the application.

3. Identify application-specific contributors

The metrics used to assess the risk impact of the application may depend on the complete PRA model or only on parts of it.

The purpose of this quantitative assessment is to identify the contributors to the risk metrics.

Ranking the contributors by their impact on the most limiting risk metric allows the further investigation in the next step into whether these contributors are potentially key sources of modeling uncertainty.

4. Assess sources of model uncertainty in the context of important contributors

The model uncertainty assessment from the base model provides insight into the model assumptions that could be related to the application specific contributors.

One option is a structured review of contributors to determine if modeling uncertainties in the PRA that affect the important contributors. This includes candidate sources of model uncertainty identified from the base PRA model assessment (via the process described in Section 3 of this report with reference to Table A-1 and Table A-2) and also includes other sources of uncertainty that were not identified from the base PRA model assessment, but that could be a source of model uncertainty for the specific application. Examples are identified in Table A-3 of this report and include items such as the following:

- Standby failure rate model
- Accident sequence modeling for level/power control in ATWS scenarios

- Treatment of rare and extremely rare events
- Credit for non-standard success paths

Tables A-1 through A-3 have been based on the expected level of detail in a base PRA model. An increased level of detail in the base model may introduce new sources of uncertainty. Table A-4 provides a list of level of detail issues that can be consulted to identify these additional sources. If any of the base PRA model candidates or the other sources of uncertainty (potentially including other items listed in Table A-4) directly affect the application important contributors, then these items are identified as sources of model uncertainty and related assumptions relevant to the application.

5. Identify sources of model uncertainty and related assumptions relevant to the application

All of the items from Step 2 or Step 4 are identified as sources of model uncertainty and related assumptions relevant to the application. These items provide input into the decision process for sensitivity study identification and the final determination of the key sources of uncertainty for the application.

4.3 Sensitivity Analyses Selection

From the items identified above, the approach to characterizing the potential key sources of uncertainty for an application makes use of a strategy that employs selected sensitivity calculations to compare with acceptance guidelines.

The general approach is to assess the potential sensitivity of the risk-informed application results to a given source of uncertainty. This accomplished by employing an alternative assumption or model and calculating the difference between the alternative approach and the baseline result for the risk metric of interest (that is, CDF or LERF). The importance and impact depend on a number of considerations that may or may not be applicable to a given source of uncertainty, as follows:

- The absolute value
- The associated probability distribution function
- The magnitude of sensitivity impact ratio
- Any correlation with other models or assumptions that may affect events in the same cutsets

The objective is to develop guidelines to assist in identifying those sources of uncertainty to which the result or application may be sensitive to the point that the related conclusion or decision may be called into question by the decision makers.

To provide a method that is consistent with the ASME/ANS PRA Standard, structured sensitivities are used as the primary decision tool. This includes the following:

- Identify and define the source of uncertainty.
  - The identification of a source of modeling uncertainty requires the recognition of the case where multiple models may exist to represent the same phenomena or physical process. This also involves defining the uncertainty in terms of an alternate logic model representation or alternate parameter values.
- Perform a sensitivity analysis to assess the impact of choosing the alternates as defined above.
- Identify if there are other epistemic uncertainties that should be coupled with this uncertainty to perform a coupled sensitivity calculation.
- Interpret the results.
- Provide the results to the decision maker in an understandable format.

For situations where a consensus model has been developed and agreed on as acceptable for a particular application, it is considered appropriate to not include this as a potential key modeling uncertainty.

4.3.1 Definition of Reasonable Range of Investigation for Sensitivity Analysis

After the potential key source of uncertainty is identified, there is a need to establish a reasonable range of parameter values or set of alternative logic models for the sensitivity evaluation. RG 1.174 provides this guidance:

> Whether the PRA is full scope or only partial scope, and whether it is only the change in metrics or both the change and baseline metrics that need to be estimated, it will be incumbent upon the licensee to demonstrate that the choice of reasonable alternative hypotheses, adjustment factors, or modeling approximations or methods to those adopted in the PRA model would not significantly change the assessment. This demonstration can take the form of well formulated sensitivity studies or qualitative arguments. In this context, "reasonable" is interpreted as implying some precedent for the alternative, such as use by other analysts, and also that there is a physically reasonable basis for the alternative. It is not the intent that the search for alternatives should be exhaustive and arbitrary.

The variation in PRA input parameters that are to be used in the sensitivity studies requires extensive judgment. In addition to basic event value changes, other sources of model uncertainty may require changes to logic model structure to perform the selected sensitivity case. Examples include changes to success criteria, incorporation of room cooling requirements for systems that do not have detailed calculations to support the lack of need for room cooling, or other changes involving system or sequence specific dependencies. In these cases, simplified representations of the alternate model logic structure may be acceptable for the sensitivity study in lieu of developing a completely different model to represent the alternative assumption. When basic event value changes are involved, this could be mathematically characterized as between the 5% and 95% bounds on the distribution. However, these distributions and bounds are often not known, and reliance on subjective assessments of such bounds is necessary.

For either type of change (logic model change or basic event value change), in some cases it may be appropriate to provide a bounding sensitivity case to demonstrate the worst possible risk metric associated with a source of uncertainty. When bounding impacts are not acceptable, however, then both increases and decreases in the risk metrics should be investigated, as appropriate. A reasonable range of variation is prescribed based on the most appropriate of the following alternatives:

- Implementation of alternate model logic
- Use of available probability distribution (if available) 5% and 95% bounds
- Use of variations identified in the literature as reasonable
- Use of judgment regarding the variations that could be expected, that is, the use of reasonable hypotheses[1]
- A factor of 2 to 10 change (in both directions, if appropriate)

Note that the intent of identifying sensitivity studies for exploration is to provide added assurance to the decision maker that a high level of confidence exists that the acceptance guidelines for the application in question are not exceeded. It is not the intent of this process to be exhaustive and arbitrary, but to define a reasonable set of sensitivity studies that might have the potential to impact the decision being made. These sensitivity studies would then be supplemented with logical combinations of sensitivity studies as applicable (refer to Section 4.3.2). The full set of sensitivity study results would be analyzed and subject to the interpretation guidance as discussed in Section 4.4.

4.3.2 Logical Combinations: Define a Reasonable Set of Sensitivity Combinations

Modeling uncertainties may create synergistic effects that have profound effects on risk metrics. Synergies have the greatest potential to impact decision makers. Therefore, there may be situations where a specific set of uncertainties represents logical combinations to be investigated.

Examples of logical combinations include the following:

- All crew actions performed under the adverse conditions of a loss of offsite ac power including the ac power restoration (adverse condition performance-shaping factor)
- Definition of core damage developed in a conservative fashion and the impact on timing used for HEPs (timing impact)
- Survivability of equipment above the EQ envelope is subject to judgment (common failure mode)

[1] Examples of the use of reasonable hypotheses for the performance of sensitivity calculations include the following:
Key failure rate: Use 95% confidence of the initiating event frequency, generic or plant-specific data.
AC power recovery: Use more conservative hypotheses on the derivation of the ac power recovery curve.
Battery life: If conservatively biased, use an increase of 50% in the battery life to represent the improved battery life.

Combinations of uncertainties are expected to result in some canceling of the effects among the different uncertainties. This assertion is based on the belief that the base model is a reasonably realistic model and that there is no significant bias in the model.

These logical combinations are to be determined using the following guidelines: 1) Multiple model uncertainties are derived from the same or related assumption, **or** 2) a specific issue related to environment, access, or specific procedural implementation affects multiple modeling uncertainties.

Additionally, the following issues were identified as potential sources of logical combinations for sensitivity studies:

- Directly related data values (i.e., appear in the same cutsets) when the values are derived from the same (or related) data pool.
- Indirectly related data values (i.e., do not appear in the same cutsets, but appear in multiple dominant contributor cutsets) when the values are derived from the same (or related) data pool.
- Synergistic scenario dependent issues (i.e., where two or more sources of model uncertainty appear in the same cutset or where different model uncertainty issues would influence the makeup of the cutset, perhaps through a revised sequence definition).

The intent of identifying logical combinations of sensitivity studies is to examine the application specific important contributors and to determine if there are related assumptions that could have an impact on the results. Again, it is not the intent of this process to be exhaustive and arbitrary, but to supplement the initial set of individual sensitivity studies with a few more sensitivity studies involving logical combinations that might have the potential to impact the decision being made. These additional sensitivity studies would be analyzed and subject to the same interpretation guidance as the individual sensitivity studies as discussed below in Section 4.4.

4.4 Interpretation of Results

Deriving an approach for interpreting the results of sensitivity studies includes recognition of the following: 1) analysis and interpretation of a large number of sensitivity cases could require significant resources, and 2) it is beneficial to define a process that screens out those contributors that have relatively small influences on the risk metrics.

The numerical decision guidelines that have been developed and published in RG 1.174, 1.177, and so on, are based on evaluations using the mean value estimates of the logic model. The development of sensitivity cases are aimed at providing additional inputs to the analysts and decision makers. There are no strict numerical acceptance guidelines developed for sensitivity calculations. The following describes guidelines that can be used in the decision-making process.

4.4.1 Sensitivity Study Result Guidelines

The overall sensitivity evaluation process for applications is captured in Figure 4-2 and the generic guidelines for evaluating sensitivity study results in the context of decision making are provided following the figure.

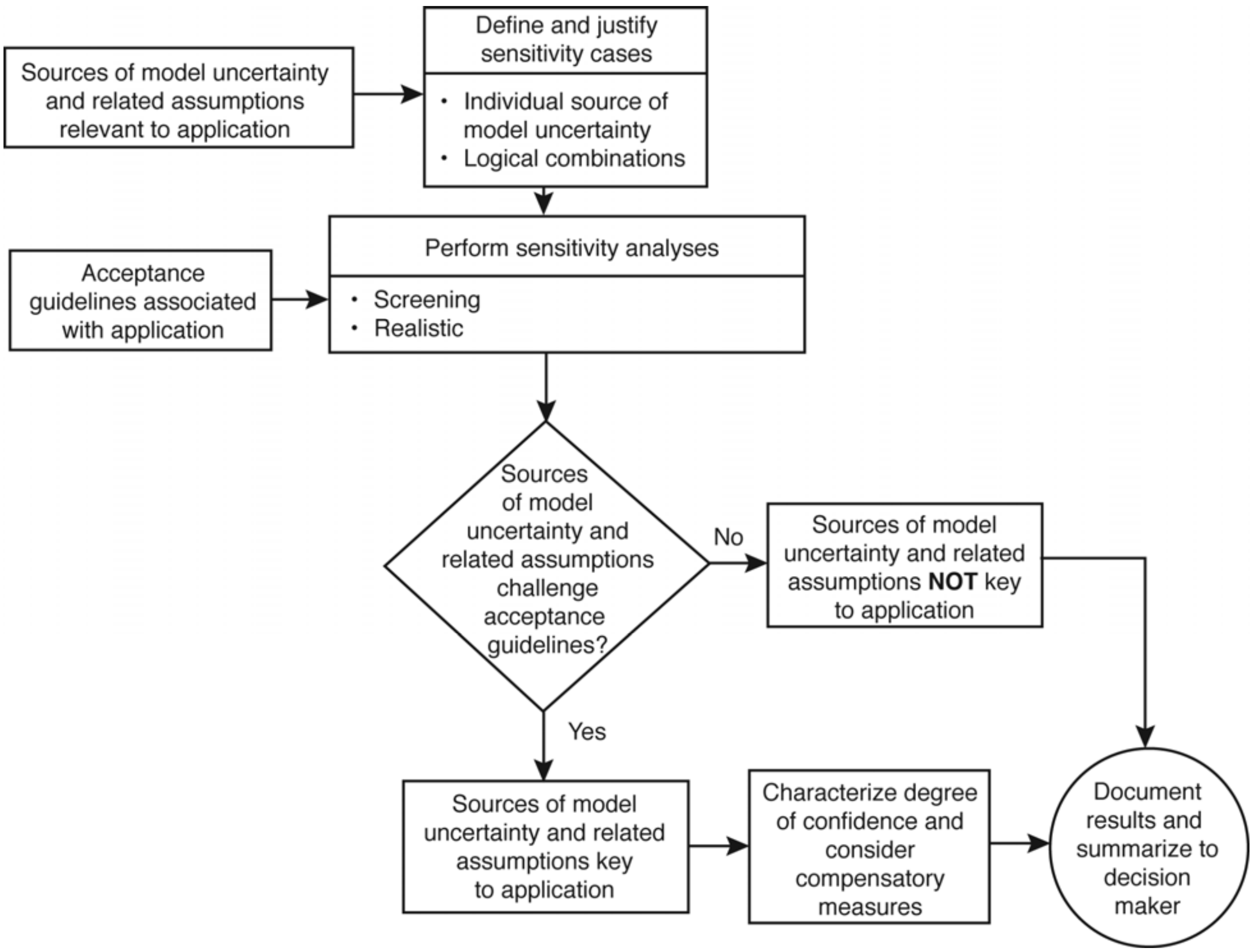

Figure 4-2
Evaluation of Sensitivity Study Impacts on Decision

Assuming that the base case results are within the acceptance guidelines for the application, then the sensitivity case results are used as the basis for assessing the significance of a source of uncertainty in the decision-making process, and when it would be useful to consider other compensatory measures that would help offset the uncertainty associated with various assumptions or model choices.

The sensitivity case must be established to represent reasonable bounds. Worst case assumptions are generally not to be used unless the worst case assumptions still clearly show that the acceptance guideline values are not threatened. In all cases, the decision maker should be provided the results and insights of the sensitivity studies.

4.4.1.1 Guideline 1

Sensitivity calculations that produce a value of the risk metric, R_S that is less than the acceptance guidelines for the application are considered *de minimis* changes and are reported to the decision maker, but are not carried forward to decision makers as a "key" source of modeling uncertainty.

4.4.1.2 Guideline 2

If sensitivity calculations produce a value of the risk metric, R_S that is greater than the acceptance guidelines, then the change introduced by the assumption calls for further characterization of the degree of confidence in the base case results to justify why the results are sufficiently described by the base case such that additional compensatory measures are not considered warranted. However, these sources of uncertainty are also reported to the decision maker and are considered as "key" sources of uncertainty.

4.4.1.3 Guideline 3

If sensitivity calculations produce a value of the risk metric, R_S that is greater than the minimum acceptance guidelines and for which sufficient justification cannot be provided to ensure that the base case results are indeed reflective of the best estimate response of the plant, then there is not a compelling argument as to why the sensitivity case can be ruled out, and as such the acceptance guideline may be challenged and these issues are also identified as "key" sources of uncertainty. In these cases, the decision maker is provided with: 1) a detailed explanation of the reason for the variation, 2) a characterization (qualitative) of the degree of confidence in the base case results, and 3) some compensatory measures that may be used to either reduce the uncertainty or reduce the resulting risk metrics.

4.5 Decision Making in the Face of Uncertainties

4.5.1 Process

Once the initial analysis is complete, utilize the guidance in Section 2.4.1 to determine if the point estimate mean or if the mean value(s) derived by the parametric uncertainty analysis should be referenced in comparison to the acceptance guidelines. Assuming that the result is within the acceptance guidelines or is otherwise deemed to be acceptable, then the initial process to identify candidate sources of model uncertainty consists of the steps outlined in Section 4.2, namely: 1) characterizing the application, 2) identifying application specific important contributors, 3) assessing the potential sources of uncertainty in the context of important contributors, and 4) identifying application-specific sources of uncertainty. Based on this initial qualitative assessment, a reasonable set of sensitivity cases for investigation is identified as outlined in Section 4.3. Finally, the results of the sensitivity studies are documented and compared with the acceptance guidelines as described in Section 4.4 so that the decision makers can be aware of the potential variations introduced by these uncertainties.

4.5.2 Decision-Maker Input

Critical in the process of understanding and effectively using the knowledge regarding key modeling uncertainties is the need to provide decision makers this information in a cogent fashion.

Expected inputs to the decision maker in the context of an application are the following:

- Provide the base risk metric quantification. If required by the application, utilize the guidance in Section 2.4.1 to determine if the point estimate value(s) or if the mean value(s) derived by the parametric uncertainty analysis should be referenced in comparison to the acceptance guidelines.
- If required by the application, also provide the parametric uncertainty bounds (estimated 5th and 95th percentile results). Utilize the guidance in Section 2.4.2 for evaluating the uncertainty interval.
- Following the guidance in Section 4.2, characterize the application and identify the candidate sources of model uncertainty for the application.
- Following the guidance in Section 4.3, identify a reasonable set of sensitivity studies to investigate and provide the results of risk metrics versus related assumptions (sensitivity cases).
- Following the guidance in Section 4.4, identify the key sources of uncertainty for the application and document them in a manner that characterizes the results of each study and the impact on the risk metrics. This characterization should explain:
  - An identification of any significant conservatisms in the modeling
  - A justification of any compensatory measures proposed to compensate for conservatisms in the model
  - A description of why any limitations of applicability are proposed
  - A description of the purpose of a proposed performance monitoring program
  - An assessment of the confidence in the recommendation

Note that NUREG-1855 (Section 7.6) provides additional guidance for what should be included in the presentation of the conclusions to the decision-maker. It is intended that the process outlined above is consistent with the expectations of NUREG-1855.

4.5.3 Example Implementation of the Process

An example hypothetical surveillance interval extension assessment for a High Pressure Coolant Injection (HPCI) Pump, Valve, and Flow Test in a BWR/4 is explored.

4.5.2.1 Characterize the Manner in which the PRA Model Is Used in the Application

The specific PRA application for which this PRA example uncertainty assessment is being developed is for the Tech Spec Initiative 5B process. This initiative allows for control of surveillance test *intervals* (STIs) to be maintained within a separate program outside of the current Technical Specifications while maintaining the surveillance test *activities* within the Technical Specifications.

The overall 5B process is a risk-informed process with the PRA model results providing one of the inputs to the Integrated Decision-making Panel (IDP) to determine if an STI change is warranted. The methodology recognizes that a key area of uncertainty for this application is the standby failure rate model utilized in the determination of the STI extension impact. Therefore, the methodology already requires the performance of selected sensitivity studies on the standby failure rate of the component(s) of interest for the STI assessment. However, the process described in this report will be followed to determine what other sources of model uncertainty are identified as potentially key for this application.

The example is a hypothetical STI assessment involving the High Pressure Coolant Injection (HPCI) Pump, Valve, and Flow Test. For the assessment, the limiting test condition is for the pump only since the valves are also fully exercised at the current test interval via the HPCI Valve Test. The total plant-specific failure probability for the HPCI pump/turbine super-component is estimated to be 1.14E-2 per demand. Without specific data available for the standby (i.e. test interval based) failure contribution, the total demand probability can be assumed time related (per the guidance in Step 8 of NEI 04-10 [10]), and the impact of doubling the test interval would be to double the total fail-to-start probability.

Therefore, the HPCI fail-to-start probability is doubled for the base case internal events PRA portion of the STI assessment. This change to the HPCI fail-to-start failure probability resulted in the base case calculated change in CDF of 3.4E-7 and the change in LERF of 2.4E-9. Both of these values are within the acceptance guidelines of the NEI 04-10 methodology.

Based on reference to Guideline 2b in Section 2.4.1, the use of the point estimate resultant values for the assessment were judged to be acceptable. That is, since the application results were determined to depend on a large number of cutsets with diverse types of contributors, not from a narrow group of state-of-knowledge correlated basic events, and the dominant contributors did not include correlated basic events. An uncertainty interval characterization of the risk metrics is also not required by the methodology guidelines for this application, and is therefore not presented to the decision makers (refer to Section 2.4.2).

4.5.2.2 Characterize Modifications to the PRA Model

In this example, there are no specific changes made to the model that introduce any application-specific sources of model uncertainty since the base PRA model is used for the application. The assessment described in Sections 4.5.2.3 and 4.5.2.4 is sufficient for identifying the sources of model uncertainty and related assumptions relevant to this application.

4.5.2.3 Identifying Application-Specific Contributors

In the example application an examination of the cutsets revealed that the following items are the important contributors to the change compared to the base case results:

- The standby failure rate values utilized for the assessment
- Operator fails to depressurize HEP values
- RCIC fails to start probability
- Turbine trip frequency, loss of feedwater, and loss of condenser vacuum initiating event frequencies
- Medium LOCA initiating event frequency
- LOOP initiating event frequency
- LOOP recovery terms at various time intervals
- Diesel generator common cause failure probabilities
- Crediting RHRSW cross-tie to ESW

4.5.2.4 Assess Sources of Model Uncertainty in Context of Important Contributors

A review of the identified sources of model uncertainty from the base model assessment as identified in Appendix B of this report was then performed to determine which of those items are potentially applicable for this assessment even though they did not appear as a dominant contributor in the base assessment for the application. Based on this review, many of the items were easily screened, but the following items were added for investigation since they were judged to be potentially applicable for this application (i.e., may have an impact on the HPCI STI extension risk assessment).

- Credit for battery life out to 4 hours without explicit representation of load shedding
- Percentage of time that two DG HVAC fans required
- Credit for core melt arrest in-vessel at high pressure
- Ex-vessel core melt progression overwhelms vapor suppression capabilities

Based on the identified important contributors from the example application of the process as described in Section 4.5.2.2 and the addition of applicable base model sources of model uncertainty, the next step is to perform a qualitative assessment to determine if sources of uncertainty have been utilized in the PRA that affect the important contributors for the application. The results of the application of this process for the HPCI surveillance test interval extension example are shown in Table 4-1.

Table 4-1
Identification of Potential Key Sources of Uncertainty

Source of Uncertainty	Source of Model Uncertainty for Base Model	Application Important Contributor	Source of Model Uncertainty Assessment	Potential Key Source of Uncertainty
Standby failure rate model	No	Yes	The calculated delta values are largely dependent upon the standby failure rates utilized for the assessment. Every cutset that contributes to the delta assessment is predicated by the absolute value change derived from the standby failure rate. Therefore, the standby failure rate is identified as a potential key source of uncertainty.	Yes
Operator fails to depressurize HEP values	Yes	Yes	An operator failure to depressurize term appears in the large majority of the cutsets that contribute to the calculated delta values. Since it is an important contributor for this application and HEP values are identified as a generic source of uncertainty for the base model, it is also retained as a potential key source of uncertainty.	Yes
RCIC fails to start probability	No	Yes	The RCIC failure probability is identified as a potential key source of uncertainty since this failure mode appears in combination in many cutsets with the HPCI fails to start probability.	Yes
Miscellaneous initiating event frequencies	No	Yes	The miscellaneous initiating event frequencies that are important in this application are not identified as potential key sources of uncertainty since their basis is well established and since their importance is only significant in combination with other identified potential key sources of uncertainty (e.g., failure to depressurize HEP values).	No

Table 4-1 (continued)
Identification of Potential Key Sources of Uncertainty

Source of Uncertainty	Source of Model Uncertainty for Base Model	Application Important Contributor	Source of Model Uncertainty Assessment	Potential Key Source of Uncertainty
LOOP Recovery Terms at Various Time Intervals	Yes	Yes	Uncertainty in the LOOP recovery probabilities will lead to some change in the calculated deltas, but the HPCI failure contribution to CDF is not limited to only LOOP events. Since there is not a first order impact on the delta assessment from the LOOP recovery values it is not independently retained as a potential key source of uncertainty, but is identified as a potential candidate for logical combinations.	Yes – in logical combinations
Diesel Generator Common Cause Failure Probabilities	No	Yes	Similarly, uncertainty in the EDG CCF probabilities will lead to some change in the calculated deltas, but the HPCI failure contribution to CDF is not limited to only LOOP events. Since there is not a first order impact on the delta assessment from the EDG CCF values it is not independently retained as a potential key source of uncertainty, but is identified as a potential candidate for logical combinations.	Yes – in logical combinations
Crediting RHRSW cross-tie to ESW	No	Yes	Although removal of credit for this success path could have some impact on the calculated deltas, it only appears in some of the cutsets that contribute to the delta. Based on this lower level second order impact, it is not retained as a potential key source of uncertainty for the HPCI pump, valve, and flow interval extension assessment.	No
Credit for battery life out to 4 hours without explicit representation of load shedding	Yes	No	A detailed review of the contributors for the assessment revealed that all of the LOOP cutsets with increased values for the HPCI fail-to-start term did not include any credit for battery life beyond two hours. That follows since the modeling only utilizes four hours of total battery life if both HPCI and RCIC succeed in the LOOP/SBO scenarios.	No

Table 4-1 (continued)
Identification of Potential Key Sources of Uncertainty

Source of Uncertainty	Source of Model Uncertainty for Base Model	Application Important Contributor	Source of Model Uncertainty Assessment	Potential Key Source of Uncertainty
Percentage of time that two DG HVAC fans required	Yes	No	The Risk Achievement Worth for this event was 1.00 for CDF and LERF in the base case assessment, so that even the maximum possible impact was assumed (i.e., 2 DG HVAC per DG cell always required), there would be no measurable change to the figures of merit.	No
Credit for core melt arrest in-vessel at high pressure	Yes	No	This source of model uncertainty does not impact CDF and is only applicable for the LERF analysis. The Risk Achievement Worth for this event was 1.00 for LERF in the base case assessment, so that even the maximum possible impact was assumed (i.e., no credit for core-melt arrest in-vessel at high pressure), there would be no measurable change to the figures of merit.	No
Ex-vessel core melt progression overwhelms vapor suppression capabilities	Yes	No	A change to the failure probabilities associated with this event could result in a first order impact on both the base model LERF values as well as the calculated deltas for the application. Therefore, the probability that ex-vessel core melt progression overwhelms vapor suppression capabilities is retained as a potential key source of uncertainty for the application.	Yes

4.5.2.5 Preparation and Presentation of Sensitivity Cases

Based on the evaluation of important contributors shown in Table 4-1, four individual sensitivity cases are identified for further exploration as potential key sources of uncertainty (i.e., sensitivity cases 1 through 4). Additionally, four logical combination sensitivity cases were identified (i.e., sensitivity cases 5 through 8). Other applications of the model may involve a few more potential key sources of uncertainty, but this is sufficient for discussion purposes here.

In preparation for discussing the sensitivity study results, it is recommended to establish the acceptance guidelines for comparison of the sensitivity calculations, and record the results. A sample presentation format for the sensitivity case results is shown in Table 4-2 for the hypothetical surveillance interval extension assessment where the minimum acceptance guideline per the NEI 04-10 [10] methodology is 1.E-6 for ΔCDF and 1E-7 for ΔLERF.

Table 4-2
Sample Presentation of Sensitivity Study Results to the Decision Maker

	ΔCDF	**ΔLERF**	**Above ΔCDF Limit?**	**Above ΔLERF Limit?**
Base Case Assessment (R_o Values)	3.9E-7	2.7E-9	No	No
Source of Uncertainty and Individual Sensitivity Study Results (R_S Values)				
1. Standby Failure Rate Model at 95th Percentile Value	2.0E-6	1.4E-8	Yes	No
2. All Operator Fails to Depressurize at 95th Percentile Value	1.1E-6 (1)	7.6E-9 (1)	Yes	No
3. RCIC Fails to Start Probability at 95th Percentile Value	5.0E-7 (1)	3.6E-9 (1)	No	No
4. Ex-vessel core melt progression overwhelms vapor suppression capabilities in high pressure core melt scenarios (i.e., guaranteed fail in high pressure scenarios) and assumed 1000x more likely in low pressure core melt scenarios (i.e., increased from 1E-5 to 1E-2).	3.9E-7 (1)	1.9E-8 (1)	No	No

Table 4-1 (continued)
Sample Presentation of Sensitivity Study Results to the Decision Maker

	ΔCDF	ΔLERF	Above ΔCDF Limit?	Above ΔLERF Limit?
Source of Uncertainty and Logical Combination Sensitivity Study Results (R_S Values)				
5. RCIC Fails to Start Probability at 95th Percentile Value and EDG CCF Values at 95th Percentile Values	5.6E-7 (1)	4.0E-9 (1)	No	No
6. RCIC Fails to Start Probability and Failure to Recover Offsite Power Values at 95th Percentile Values	6.0E-7 (1)	4.4E-9 (1)	No	No
7. RCIC Fails to Start Probability, EDG CCF Values, and Failure to Recover Offsite Power Values at 95th Percentile Values	7.4E-7 (1)	5.4E-9 (1)	No	No
8. RCIC Fails to Start Probability and All Operator Fails to Depressurize at 95th Percentile Value	1.4E-6 (1)	1.1E-8 (1)	Yes	No

(1) To isolate the ΔCDF and ΔLERF values related to the surveillance test interval extension, revised base CDF and LERF values are established first and then the change associated with the STI change is determined and reported here.

A review of the sensitivity case results in Table 4-2 indicates that the threshold acceptance value is exceeded in the first two sensitivity cases for ΔCDF. As such the standby failure rate used for the assessment and the operator error terms for failing to depressurize are identified as a key sources of uncertainty for the application. The results of the third sensitivity case provides confidence that even if the RCIC fail to start failure probability is much higher than the current PRA model employs, then the acceptance guidelines for the application are not exceeded. Additionally, the results of even the very pessimistic assumptions for ex-vessel core melt progression in sensitivity case 4 are not much worse than the first sensitivity case ΔLERF result.

Sensitivity Cases 5, 6, and 7 represent logical combinations involving synergistic impacts for LOOP events (i.e., various combinations for RCIC, EDG CCF, and offsite power recovery probabilities). The results indicate that even with all of the related parameters at reasonable upper bound values, the acceptance guidelines would not be exceeded. Sensitivity Case 8 shows that if the RCIC fail to start probability is set to a reasonable upper bound value along with the fail to depressurize HEPs at their 95th percentile values as well, then the acceptance guideline for CDF is further exceeded than if only the fail to depressurize HEPs are adjusted to higher values. Given that the fail to depressurize HEP case (Case 2) was already above the acceptance guidelines, however, this does not indicate that RCIC should also be identified as a key source of uncertainty but rather further confirms the premise that the operator terms for failing to depressurize the are a key source of uncertainty for the application.

In summary, the following two items are identified as key sources of uncertainty for this application of the PRA model:

- HPCI standby failure rate model
- Failure to depressurize RPV Human Error Probability values

Following the guidance in Section 4.4, since the standby failure rate model and fail to depressurize HEPs are identified as key sources of uncertainty, it would be incumbent upon the analyst to characterize the degree of confidence in the derived HEP values and the assumptions associated with the standby failure rate model that lead to the base case results being within the acceptance guidelines. This would likely include evidence of using accepted HRA methodologies with a strong basis for the assumed times available for the fail to depressurize HEP source of uncertainty. Additionally, evidence that the use of the linear standby failure rate model with all failures assumed to be time related and not accounting for shock related failure contribution is adequately conservative for this assessment (e.g., by referencing satisfactory operating experience of similar components at the extended test interval that justifies the use of a linear model compared to a non-linear model for the standby failure model) would help to support the degree of confidence in the base case result. On the other hand, other compensatory measures (e.g., enhanced performance monitoring) might need to be identified to present to the decision maker to provide assurance that the identified key source of uncertainty is not adversely affecting the decision.

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A
GENERIC SOURCES OF MODELING UNCERTAINTY

A.1 Overview

The grouping chosen for this analysis are generally consistent with the literature regarding uncertainty treatment, for example, RG 1.174 [4] where the three high-level types of uncertainties are the following:

- Parametric uncertainty
- Modeling uncertainty
- Completeness uncertainty

The focus of this appendix is to provide a discussion related to those items that have been earmarked as candidate sources of modeling uncertainty.

A generic list of potential model uncertainties is developed through the examination of the ASME/ANS PRA Standard and available industry/NRC PRAs (see Appendix H of the Technical Basis Document). Upon development of the initial broad list and based on interactions with NRC personnel, it became apparent that most of the items are related to scope or level of detail issues that can relate to simplifying assumptions rather than true modeling uncertainty issues. True modeling uncertainty issues would lead to assumptions that are made with the knowledge that a different reasonable alternative assumption exists. Alternatively, an assumption related to scope or level of detail is simply one that is made for modeling convenience.

The candidate sources of modeling uncertainty list has been developed with the intent of defining that set of issues that need to be addressed to satisfy the following supporting requirements from the ASME/ANS PRA Standard [26].

- QU-E1: IDENTIFY sources of model uncertainty.
- QU-E2: IDENTIFY assumptions made in the development of the PRA model.
- QU-E4: For each source of model uncertainty and related assumption identified in QU-E1 and QU-E2, respectively, IDENTIFY how the PRA model is affected (e.g., introduction of a new basic event, changes to basic event probabilities, change in success criterion, introduction of a new initiating event)
- QU-F4: DOCUMENT the characterization of the sources of model uncertainty and related assumptions (as identified in QU-E4).

- LE-F3: IDENTIFY and CHARACTERIZE the LERF sources of model uncertainty and related assumptions, consistent with the requirements of Tables 2.2.7-2(d) and 2.2.7-2(e).
- IE-D3, AS-C3, SC-C3, SY-C3, HR-I3, DA-E3, LE-G4, and IF*-B3: DOCUMENT the sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2 [or LE-F3]) associated with …[each element].

A.2 Identification of Sources of Uncertainty

In the process of identifying and characterizing assumptions associated with the potential sources of modeling uncertainty, it is important to establish a definition for such assumptions. This definition has also been provided by the NRC via NUREG-1855 [3]:

> *An assumption related to a model uncertainty* is either generally accepted, or is made with the knowledge that a different reasonable alternative assumption exists. *A reasonable alternative assumption* is one that has broad acceptance within the technical community and for which the technical basis for consideration is at least as sound as that of the assumption being made.

Note that reasonable alternative assumptions can lead to increases or decreases in the calculated risk metrics, and the process should recognize that both outcomes are possible. The development of the base PRA model therefore requires a careful balance of conservative bias treatment with realistic assessments to ensure that the base PRA model risk profile is adequately developed. Applications of the PRA model could become more or less dependent on the choices made for addressing modeling uncertainty. The intent is to generically define those set of modeling uncertainty issues that need to be identified and characterized to meet the supporting requirement definitions provided above.

As such, the following definition for identifying modeling uncertainty issues has been developed:

- Significant interpretations to infer behavior are required to develop a model (this is the case where some information is available, but is not sufficient to derive a definitive model or value),
- The phenomena or nature of the event or failure mode being modeled is not completely understood, or
- There is general agreement that the issue represents a potential source of modeling uncertainty.

The process for identifying and characterizing the interpreted and phenomenological sources of uncertainty is described in Section A.2.1. A slightly different approach is taken for the other model uncertainty issues as described in Section A.2.2. Section A.2.3 provides a supplementary list to those items appearing in Section A.2.1. These issues were identified as examples of potential sources of model uncertainty that may have impacts on specific applications of the model but will typically not be significant contributors to the base model assessment. As such, plant-specific identification and characterization of these issues would not be required to meet

Capability Category II of the ASME/ANS PRA Standard, but are included here to provide examples (i.e., it is not an exhaustive and comprehensive list) of other items to consider when performing applications.

A.2.1 Addressing Interpreted and Phenomenological Modeling Uncertainty Issues

Section 3.2 describes the process of how the items that originally appeared in Appendix H of the Technical Basis Document [1] were reviewed and evolved into the identification of those issues that have been earmarked as modeling uncertainty candidates based on the definitions provided above. Given that the candidate model uncertainty list exists, to meet the PRA Standard Supporting Requirements defined above for QU-E1, QU-E2, QU-E4, QU-F4, and LE-F3 a plant-specific issue characterization needs to be provided. This could take the form shown in Figure A-1 below.

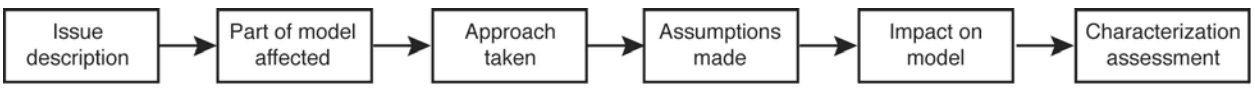

Figure A-1
Identifying and Characterizing Model Uncertainty Issues

To assist with this process, Figure A-2 provides an overview for the issue characterization that is provided for each item in this Appendix. That is, for each model uncertainty issue that has been identified, an indication is provided for: (1) a discussion of the issue, (2) typical parts of the model affected, and (3) representative sample approaches. These three aspects are included in Table A-1 where the impact on the model and the characterization assessment is left to be uniquely defined for each model. Note that the typical approaches listed in Table A-1 are provided as examples of how one might characterize the potential sources of model uncertainty for their model. The inclusion of the approach in the list does not imply that it is a preferred or endorsed approach, nor does the use of any of the approaches provide a means to screen the source of uncertainty for consideration as a potential key source of uncertainty in applications.

The intent of Table A-1 is to identify the minimum set of generic sources of model uncertainty that need to be addressed for meeting the QU-E1, QU-E2, QU-E4, QU-F4, and LE-F3 and other related PRA Standard supporting requirements defined above, and to provide a template for the approach to be followed to do so. That is, for each applicable item in Table A-1, a plant-specific issue characterization should be provided to fully satisfy the related supporting requirements. In general, information from one of the typical approaches described here could be applicable to the plant-specific issue characterization development.

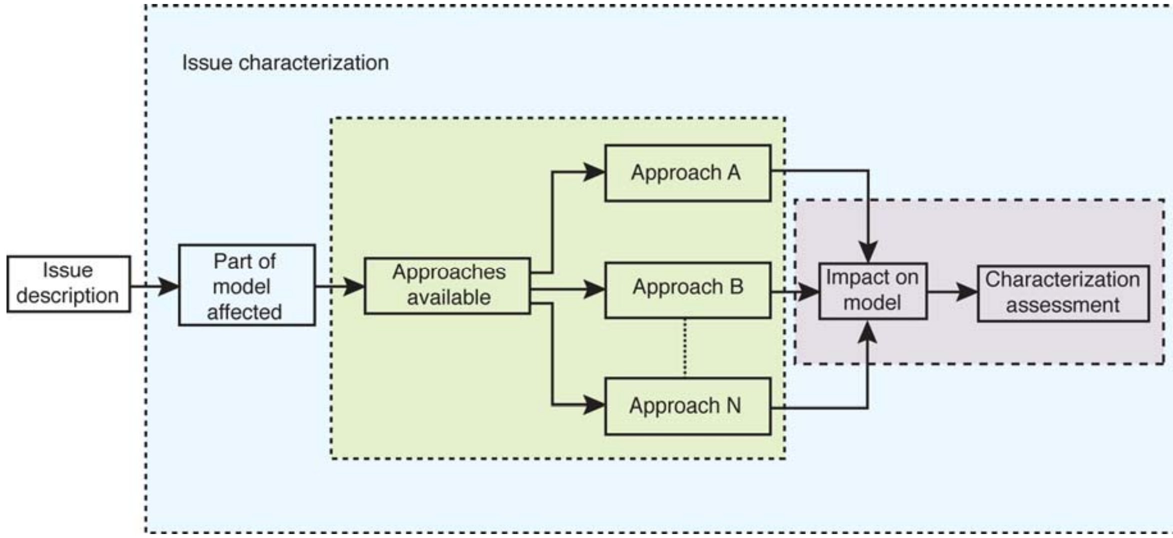

Figure A-2
Template for Model Uncertainty Issue Characterization

Also note that the list of possible approaches is not exhaustive, and its inclusion on this list is not an indication that the approach would represent a consensus model or would represent an acceptable conservative bias approach. The possible approaches are provided as examples, but in any event the plant-specific issue characterization would need to be provided and the impact on the model would need to be uniquely defined and assessed as candidate sources of model uncertainty per the guidance provided in Section 3.1.1 of this report. A full example implementation of the process for each of the issues identified in Table A-1 is provided in Appendix B of this report.

Table A-1
Issue Characterization for Sources of Model Uncertainty

Issue Description		Issue Characterization	
Topic	**Discussion of Issue**	**Part of Model Affected**	**Possible Approaches (Not Exhaustive)**
Initiating Event Analysis (IE)			
1. Grid stability	The LOOP frequency is a function of several factors including switchyard design, the number and independence of offsite power feeds, the local power production and consumption environment and the degree of plant control of the local grid and grid maintenance. Three different aspects relate to this issue: 1a. LOOP initiating event frequency values and recovery probabilities 1b. Conditional LOOP probability 1c. Availability of dc power to perform restoration actions	LOOP sequences	1. LOOP frequencies for the different categories and recovery probabilities based on data from NUREG/CR-6890 [11]. Each LOOP category uniquely represented in the model.
			2. Different categories merged into single LOOP frequency with weighted average recovery probabilities.
			3. Update of data for recent experience and use separate analysis to account for plant-specific or regional grid stability issues.
		Consequential LOOP sequences	4. Conditional LOOP frequencies based on NRC recommended values [12, 20].
			5. Conditional LOOP frequencies based on EPRI expert elicitation values [13] or Owners Group assessments.
		LOOP or consequential LOOP sequences with offsite power recovered	6. Plant-specific features and dependencies accounted for in system modeling of LOOP restoration after ac power recovery occurs.
			7. Use of generic data assumed to adequately account for availability of dc power to perform restoration actions.

Table A-1 (continued)
Issue Characterization for Sources of Model Uncertainty

Issue Description		Issue Characterization	
Topic	**Discussion of Issue**	**Part of Model Affected**	**Possible Approaches (Not Exhaustive)**
2. Support System Initiating Events	Increasing use of plant-specific models for support system initiators (e.g., loss of SW, CCW, or IA, and loss of ac or dc buses) have led to inconsistencies in approaches across the industry. A number of challenges exist in modeling of support system initiating events: 2a. Treatment of common cause failures 2b. Potential for recovery	Support system event sequences	1. Use annualized rates for common cause failures of normally operating components in a detailed SSIE model with extensive fault tree and plant specific inputs.
			2. Screen CCF events to create new CCF factors to address potential for repair/realignment
		Support system event sequences	3. Assume NUREG/CR-6928 [14] value, but credit recovery based on scenario-specific attributes.
			4. Utilize support system initiating event frequencies as derived with no additional credit for recovery.
3. LOCA initiating event frequencies	It is difficult to establish values for events that have never occurred or have rarely occurred with a high level of confidence. The choice of available data sets or use of specific methodologies in the determination of LOCA frequencies could impact base model results and some applications.	LOCA sequences	1. Base LOCA frequencies on plant-specific pipe segment count from EPRI methodology [15].
			2. Utilize the data provided in NUREG/CR-6928 [14] or other acceptable reference.
			3. Utilize data referenced in NUREG/CR-6928 [14], but confirm applicability for plant-specific implementation from original data source in NUREG-1829 [19].

Table A-1 (continued)
Issue Characterization for Sources of Model Uncertainty

Issue Description		Issue Characterization	
Topic	**Discussion of Issue**	**Part of Model Affected**	**Possible Approaches (Not Exhaustive)**
Accident Sequence Analysis (AS)			
4. Operation of equipment after battery depletion	Station Blackout events are important contributors to baseline CDF at nearly every US NPP. In many cases, battery depletion may be assumed to lead to loss of all system capability. Some PRAs have credited manual operation of systems that normally require dc for successful operation (e.g., turbine-driven systems such as RCIC and AFW).	Credit for continued operation of these systems in sequences with batteries depleted (e.g., long-term SBO sequences)	1. Assume failure of dc powered SSCs upon battery depletion.
			2. Credit manual operation of selected systems using screening HEP following battery depletion.
			3. Limited credit for manual operation based on specific scenarios where HRA Performance Shaping Factors can be satisfied.
			4. Credit for manual operation of systems leads to extended time available, but does not lead to a complete success path.
5. RCP seal LOCA treatment – PWRs	The assumed timing and magnitude of RCP seal LOCAs given a loss of seal cooling can have a substantial influence on the risk profile.	Accident sequences involving loss of seal cooling	1. Utilize PWROG Seal LOCA consensus model approach for Westinghouse [16] or CE plants [17].
			2. Provide sufficient justification for seal LOCA treatment (alternative timing and sizes assumed) or provide plant-specific features of why seal LOCAs should not occur.

Table A-1 (continued)
Issue Characterization for Sources of Model Uncertainty

Issue Description		Issue Characterization	
Topic	**Discussion of Issue**	**Part of Model Affected**	**Possible Approaches (Not Exhaustive)**
6. Recirculation pump seal leakage treatment – BWRs w/ Isolation Condensers	Recirculation pump seal leakage can lead to loss of the Isolation Condenser. While recirculation pump seal leakage is generally modeled, there is no consensus approach on the likelihood of such leaks.	Accident sequences with long-term use of isolation condenser	1. No credit for IC long-term due to concerns about seal leakage
			2. Single leak size assumed. Recovery of injection and/or alternate cooling required prior to core damage.
			3. Distribution of leak sizes assumed. Recovery of injection and/or alternate cooling required prior to core damage, depending on leak.
Success Criteria (SC)			
7. Impact of containment venting on core cooling system NPSH	Many BWR core cooling systems utilize the suppression pool as a water source. Venting of containment as a decay heat removal mechanism can substantially reduce NPSH, even lead to flashing of the pool. The treatment of such scenarios varies across BWR PRAs.	Loss of containment heat removal scenarios with containment venting successful	1. No credit for injection from suppression pool following venting.
			2. HFE defined and incorporated into PRA for control of containment pressure in order to assure adequate NPSH.
			3. Analysis developed to demonstrate continued injection, despite reduction in NPSH.
			4. Injection from suppression pool assumed to be unaffected by venting.

Table A-1 (continued)
Issue Characterization for Sources of Model Uncertainty

Issue Description		Issue Characterization	
Topic	**Discussion of Issue**	**Part of Model Affected**	**Possible Approaches (Not Exhaustive)**
8. Core cooling success following containment failure or venting through non hard pipe vent paths	Loss of containment heat removal leading to long-term containment over-pressurization and failure can be a significant contributor in some PRAs. Consideration of the containment failure mode might result in additional mechanical failures of credited systems. Containment venting through "soft" ducts or containment failure can result in loss of core cooling due to environmental impacts on equipment in the reactor/auxiliary building, loss of NPSH on ECCS pumps, steam binding of ECCS pumps, or damage to injection piping or valves. There is no definitive reference on the proper treatment of these issues.	Long term loss of decay heat removal sequences	1. All core cooling systems assumed to be failed in all scenarios involving containment failure and venting.
			2. Selected core cooling systems assumed to be unaffected in specific scenarios involving containment failure and venting.
			3. Core cooling systems in the Reactor/Auxiliary Building assumed to be unaffected in all scenarios involving containment failure and venting.
			4. Analysis performed to demonstrate continued viability of selected systems post-venting or post containment failure.

Table A-1 (continued)
Issue Characterization for Sources of Model Uncertainty

Issue Description		Issue Characterization	
Topic	**Discussion of Issue**	**Part of Model Affected**	**Possible Approaches (Not Exhaustive)**
9. Room heatup calculations	Loss of HVAC can result in room temperatures exceeding equipment qualification limits. Treatment of HVAC requirements varies across the industry and often varies within a PRA. There are two aspects to this issue. One involves whether the SSCs affected by loss of HVAC are assumed to fail (i.e., there is uncertainty in the fragility of the components). The other involves how the rate of room heatup is calculated and the assumed timing of the failure.	Dependency on HVAC for system modeling and timing of accident progressions and associated success criteria.	1. Assume loss of design basis HVAC leads to a loss of SSC function at t=0.
			2. Analysis developed to show that inadequate HVAC (based on realistic assessment) leads to an early loss of SSC function (i.e., at t=0).
			3. Analysis developed to show that inadequate HVAC (based on realistic assessment) leads to a delayed loss of SSC function at some point in time based on analysis.
			4. Analysis developed to show that inadequate HVAC (based on realistic assessment) does not lead to loss of SSC function within the mission time for system operation.

Table A-1 (continued)
Issue Characterization for Sources of Model Uncertainty

Issue Description		Issue Characterization	
Topic	**Discussion of Issue**	**Part of Model Affected**	**Possible Approaches (Not Exhaustive)**
10. Battery life calculations	Station Blackout events are important contributors to baseline CDF at nearly every US NPP. Battery life is an important factor in assessing a plant's ability to cope with an SBO. Many plants only have Design Basis calculations for battery life. Other plants have very plant/condition-specific calculations of battery life. Failing to fully credit battery capability can overstate risks, and mask other potential contributors and insights. Realistically assessing battery life can be complex.	Determination of battery depletion time(s) and the associated accident sequence timing and related success criteria.	1. Use design basis battery life. 2. Use plant-specific battery life based on bounding of expected loads and battery conditions. 3. Use plant-specific battery life based on realistic assessment of expected loads and battery conditions.
11. Number of PORVs required for bleed and feed – PWRs	PWR EOPs direct opening of all PORVs to reduce RCS pressure for initiation of bleed and feed cooling. Some plants have performed plant-specific analysis that demonstrate that less than all PORVs may be sufficient, depending on ECCS characteristics and initiation timing.	System logic modeling representing success criterion and accident sequence timing for performance of bleed and feed and sequences involving success or failure of feed and bleed.	1. Assume all PORVs required for bleed and feed success. 2. Number of PORVs required based on plant-specific thermal hydraulic analysis. 3. Variable success criteria incorporated into model for initiating event and/or scenario-specific differences.

Table A-1 (continued)
Issue Characterization for Sources of Model Uncertainty

Issue Description		Issue Characterization	
Topic	**Discussion of Issue**	**Part of Model Affected**	**Possible Approaches (Not Exhaustive)**
12. Containment sump / strainer performance	All PWRs are improving ECCS sump management practices, including installation of new sump strainers at most plants. All BWRs have improved their suppression pool strainers to reduce the potential for plugging. However, there is not a consistent method for the treatment of suppression pool strainer performance.	Recirculation from sump (PWRs) or from the suppression pool (BWRs) system modeling and sequences involving injection from these sources (Note that the modeling should be relatively straightforward, the uncertainty is related to the methods or references used to determine the likelihood of plugging the sump strainer and common cause failure by blockage of the strainers.)	1. Provide analysis for the probability of common cause failure by blockage of sump strainers/suppression pool strainers. 2. Utilize PWROG sump model event tree and branch quantification guidance for the treatment of ECCS sump performance. 3. Assume no common cause potential.
13. Impact of failure of pressure relief	Certain scenarios can lead to RCS/RPV pressure transients requiring pressure relief. Usually, there is sufficient capacity to accommodate the pressure transient. However, in some scenarios, failure of adequate pressure relief can be a consideration. Various assumptions can be taken on the impact of inadequate pressure relief.	Success criterion for prevention of RPV overpressure (Note that uncertainty exists in both the determination of the global CCF values that may lead to RPV overpressure and what is done with the subsequent RPV overpressure sequence modeling.)	1. Assume failure to provide adequate pressure relief leads to core damage. 2. Assume that failure to provide adequate pressure relief leads to a large LOCA. 3. Assume that failure to provide adequate pressure relief leads to failure of an SRV pipe. 4. Assume that failure to provide adequate pressure relief has no impact or has negligible probability.

Table A-1 (continued)
Issue Characterization for Sources of Model Uncertainty

Issue Description		Issue Characterization	
Topic	**Discussion of Issue**	**Part of Model Affected**	**Possible Approaches (Not Exhaustive)**
Systems Analysis (SY)			
14. Operability of equipment in beyond design basis environments	Due to the scope of PRAs, scenarios may arise where equipment is exposed to beyond design basis environments (w/o room cooling, w/o component cooling, w/ deadheading, in the presence of an unisolated LOCA in the area, etc.).	System and accident sequence modeling of available systems and required support systems	1. Assume beyond design basis condition leads to loss of SSC function at t=0.
			2. Assume that beyond design basis condition (based on realistic analysis) leads to loss of SSC function at a point in time based on analysis.
			3. Assume beyond design basis condition does not lead to loss of SSC function.
Human Reliability Analysis (HR)			
15. Credit For ERO	Most PRAs do not give much, if any credit, for initiation of the Emergency response Organization (ERO), including actions included in plant-specific SAMGs and the new B5b mitigation strategies. The additional resources and capabilities brought to bear via the ERO can be substantial, especially for long-term events.	System or accident sequence modeling with incorporation of HFEs and HEP value determination in both the Level 1 and Level 2 models	1. No credit given to ERO.
			2. Credit for selected SAMGs implemented via the ERO.
			3. All applicable ERO capabilities credited.

Table A-1 (continued)
Issue Characterization for Sources of Model Uncertainty

Issue Description		Issue Characterization	
Topic	**Discussion of Issue**	**Part of Model Affected**	**Possible Approaches (Not Exhaustive)**
Internal Flooding (IF)			
16. Piping failure mode	One of the most important, and uncertain, inputs to an internal flooding analysis is the frequency of floods of various magnitudes (e.g., small, large, catastrophic) from various sources (e.g., clean water, untreated water, salt water, etc.). EPRI has developed some data, but the NRC has not formally endorsed its use.	Likelihood and characterization of internal flooding sources and internal flood event sequences and the timing associated with human actions involved in flooding mitigation	1. Maximum flow rate assumed in all cases based on system design characteristics.
			2. Maximum flow rate reduced based on location-specific or scenario-specific characteristics.
			3. Multiple flow rate scenarios assumed for each flood location.
LERF Analysis (LE)			
17. Core melt arrest in-vessel	Typically, the treatment of core melt arrest in-vessel has been limited. However, recent NRC work has indicated that there may be more potential than previously credited. An example is credit for CRD in BWRs.	LERF / Level 2 containment event tree sequences	1. No credit for arresting core melt in-vessel.
			2. Credit for arresting in-vessel only in selected scenarios (e.g., upon recovery of offsite power)
			3. Credit for arresting core melt in-vessel based on scenario-specific assessment.

Table A-1 (continued)
Issue Characterization for Sources of Model Uncertainty

Issue Description		Issue Characterization	
Topic	**Discussion of Issue**	**Part of Model Affected**	**Possible Approaches (Not Exhaustive)**
18. Thermally induced failure of hot leg/SG tubes – PWRs	NRC analytical models and research findings continue to show that a thermally induced steam generator tube rupture (TI-SGTR) is more probable than predicted by the industry. There is a need to come to agreement with NRC on the thermal hydraulics modeling of TI SGTR.	LERF / Level 2 containment event tree sequences	1. TI-SGTR assumed to occur in all high RCS pressure, dry steam generator, low RCS level (hi-dry-low) scenarios.
			2. Assume probability of TI-SGTR in all hi-dry-low scenarios.
			3. TI-SGTR assumed to never occur based on detailed thermal hydraulic analysis.
19. Vessel failure mode	The progression of core melt to the point of vessel failure remains uncertain. Some codes (MELCOR) predict that even vessels with lower head penetrations will remain intact until the water has evaporated from above the relocated core debris. Other codes (MAAP), predict that lower head penetrations might fail early. The failure mode of the vessel and associate timing can impact LERF binning, and may influence HPME characteristics (especially for some BWRs and PWR ice condenser plants).	LERF / Level 2 containment event tree sequences	1. Assume vessel fails due to local penetration failure.
			2. Assume vessel fails after dryout of lower head.
			3. Vessel failure timing based on mechanistic assessment of vessel heatup.

Table A-1 (continued)
Issue Characterization for Sources of Model Uncertainty

Issue Description		Issue Characterization	
Topic	**Discussion of Issue**	**Part of Model Affected**	**Possible Approaches (Not Exhaustive)**
20. Ex-vessel cooling of lower head	The lower vessel head of some plants may be submerged in water prior to the relocation of core debris to the lower head. This presents the potential for the core debris to be retained in-vessel by ex-vessel cooling. This is a complex analysis impacted by insulation, vessel design and degree of submergence.	LERF / Level 2 containment event tree sequences	1. Ex-vessel cooling of the lower head not possible due to design of the plant.
			2. Ex-vessel cooling of the lower head possible but not credited as a heat removal mechanism.
			3. Ex-vessel cooling of the lower head credited in certain scenarios as a means of retaining an integral lower head.
21. Core debris contact with containment	In some plants, core debris can come in contact with the containment shell (e.g., some BWR Mark Is, some PWRs including free-standing steel containments). Molten core debris can challenge the integrity of the containment boundary. Some analyses have demonstrated that core debris can be cooled by overlying water pools.	LERF / Level 2 containment event tree sequences	1. Containment barrier not susceptible to damage by debris.
			2. Containment failure assumed any time debris contacts containment shell.
			3. Under certain conditions, containment failure assumed when debris contacts containment shell.

Table A-1 (continued)
Issue Characterization for Sources of Model Uncertainty

Issue Description		Issue Characterization	
Topic	**Discussion of Issue**	**Part of Model Affected**	**Possible Approaches (Not Exhaustive)**
22. ISLOCA IE Frequency Determination	ISLOCA is often a significant contributor to LERF. One key input to the ISLOCA analysis are the assumptions related to common cause failure of isolation valves between the RCS/RPV and low pressure piping. There is no consensus approach to the data or treatment of this issue. Additionally, given an overpressure condition in low pressure piping, there is uncertainty surrounding the failure mode of the piping.	ISLOCA initiating event sequences	1. ISLOCA development includes the relevant considerations listed in IE-C12 of the ASME/ANS PRA Standard and accounts for common cause failures and captures likelihood of different piping failure modes.
			2. ISLOCA development based on generic assessments.
			3. ISLOCAs screened from model as extremely low likelihood events.
23. Treatment of Hydrogen combustion in BWR Mark III and PWR ice condenser plants	The amount of hydrogen burned, the rate at which it is generated and burned, the pressure reduction mitigation credited by the suppression pool, ice condenser, structures, etc. can have a significant impact on the accident sequence progression development.	Level 2 containment event tree sequences	1. The amount of hydrogen generated and subsequent impact of burns based on plant-specific and sequence-specific MAAP thermal/hydraulic calculations.
			2. Several plant-specific sensitivity cases explored to determine the potential impacts of hydrogen combustion with the results of the sensitivity cases factored into the probabilistic evaluation.
			3. Plant-specific calculations performed to provide an upper bound on the potential impact of hydrogen production and impact from combustion.

A.2.2 Addressing Other Generic Modeling Uncertainty Issues

Table A-2 provides a similar framework as Table A-1 for three additional sources of model uncertainty that were not identified from a phenomenological or interpreted behavior perspective. Rather, these three issues were identified as potential sources of model uncertainty since they are generally understood and accepted as areas of uncertainty that can be significant contributors to CDF and LERF.

As part of the lessons learned in the original invocation of the process outlined in the Application Guide [2], a recommendation was made to include the development of a standard set of sensitivity cases to perform that may envelope several potential sources of uncertainty at a relatively high level. The process of developing a set of standard sensitivity cases has precedence in NEI 00-04 [18]. Developing the set of sensitivity cases in lieu of trying to identify and characterize all potential sources of uncertainty associated with these issues has the potential benefit of highlighting the potential impact of these specific issues prior to performing applications. Therefore, a standard set of four sensitivity cases is recommended as follows:

- All HEP probabilities (including pre-initiators, post-initiators, and dependent HEP values) set to their 5th percentile value (the use of zero-value HEP probabilities is also deemed acceptable)
- All HEP probabilities (including pre-initiators, post-initiators, and dependent HEP values) set to their 95th percentile value
- All CCF probabilities set to their 5th percentile value (the use of zero-value CCF probabilities is also deemed acceptable)
- All CCF probabilities set the their 95th percentile value

The results of these analyses can be compared to the RG 1.174 CDF and LERF limits of $1x10^{-4}$/yr for CDF and $1x10^{-5}$/yr for LERF to obtain insights into the sensitivity of the base PRA model results to these generic high level sources of modeling uncertainty.

The presentation of the results of the sensitivity cases described above would provide added assurance of model understanding prior to performing applications. Individual applications of the model would then need to consider if specific Human Failure Events (HFEs), dependent HFEs, or common cause failure (CCF) events have a cause-effect relationship in the context of important contributors for the application. Following the process outlined in Section 4 of this report, these items might become candidates for sensitivity studies. However, it is also recognized that using accepted best practices for the derivation of these values and meeting Capability Category II or better of the ASME/ANS PRA Standard [26] for these items provides a high degree of confidence in the acceptability of the base case values for these events. The characterization of the degree of confidence in the values is also accounted for in the process outlined in Section 4. Additionally, variations in the parameters could also be unreasonable due to failure to account for compensatory measures (e.g., the performance of pre-shift briefs by operations staff to ensure high reliability on the execution of those HFEs that are determined to be key sources of uncertainty for a specific application).

Table A-2
Issue Characterization for Other Sources of Model Uncertainty

Issue Description		Issue Characterization	
Topic	**Discussion of Issue**	**Part of Model Affected**	**Possible Approaches (Not Exhaustive)**
Human Reliability Analysis (HR)			
24. Basis for HEPs	There is not a consistent method for the treatment of pre-initiator and post-initiator human errors. However, human failures events are typically significant contributors to CDF and LERF.	System or accident sequence modeling with incorporation of HFEs and HEP value determination	1. Screening HEPs based on an accepted method used for many or all HFEs.
			2. Realistic HEPs based on an accepted method used for significant HFEs.
			3. Realistic HEPs based on an accepted method used for all HFEs.
Quantification (QU)			
25. Treatment of HFE dependencies	There is not a consistent method for the treatment of potentially dependent post-initiator human errors. SPAR models do not generally include dependencies.	Quantification of dependent human errors	1. No dependence applied.
			2. Some HFEs identified as dependent and with the dependent HFEs incorporated into the model directly or accounted for with post-processing recovery rules.
			3. All HFEs in same cutset assessed for dependence and with the dependent HFEs incorporated into the model directly or accounted for with post-processing recovery rules.
Data Analysis (DA)			
26. Intra-system common cause events	Common cause failures have been shown to be important contributors in PRAs. As limited plant-specific data is available, generic common cause factors are commonly used. Sometimes, plant-specific evidence can indicate that the generic values are inappropriate.	CCF data values and associated system model representations	1. Generic CCF factors used (α factor, β factor, or MGL factors).
			2. Plant-specific CCF factors generated based on event screening.

A.2.3 Identification of Other Potential Modeling Uncertainty Issues

Table A-3 provides a supplementary list to those items appearing in Section A.2.1. These issues were identified as examples of potential sources of model uncertainty that may have impacts on specific applications of the model but will typically not be significant contributors to the base model assessment. As such, plant-specific identification and characterization of these issues as specific sources of model uncertainty would not be required to meet Capability Category II of the ASME/ANS PRA Standard for those SRs identified above in Section A.1. This list is not exhaustive and comprehensive, but is included here to provide examples of other model uncertainty items that may be important to consider when performing applications. Additionally, it is expected that specific shortcomings for many of these potential sources of model uncertainty would have been identified as part of the peer review process.

Table A-3
Supplementary List for Other Potential Sources of Model Uncertainty

Topic	Discussion of Issue
1. Treatment of boron dilution events.	The treatment of boron dilution in PWRs may vary significantly in both the deterministic models and in the quantitative probabilistic model. Some applications of the model might be driven by the specific treatment chosen. Typically these events proceed slowly and an will be prevented prior to becoming a potential core damage scenario.
2. Selection of prior distributions when carrying out a Bayesian analysis of data.	In general, the peer review should be adequate to ensure that reasonable prior values are employed. However, some applications might be driven by uncertainty in the selection of the prior distributions utilized for either initiating event frequency development or in the component failure data development.
3. Treatment of rare and extremely rare events.	In general, the peer review should be adequate to ensure that reasonable values are employed. Selection of data should be based on confirmation that the database used is applicable to the plant (e.g., no unique failure modes not considered in the data base are active).
4. Moderator temperature coefficient – important in PWR ATWS.	The peer review should address the basis for the MTC used in the PRA based on plant specific data. This could be an application-specific uncertainty, however, especially for an application focused around ATWS mitigation.
5. Pressurized Thermal Shock – PWRs.	Consideration of pressurized thermal shock after the secondary system is depressurized, for example, after a main steam line break can lead to sources of model uncertainty.

Table A-3 (continued)
Supplementary List for Other Potential Sources of Model Uncertainty

Topic	Discussion of Issue
6. Credit for non-standard success paths (e.g., use of alternate injection systems).	In general, the peer review should be adequate to ensure that unreasonable credit for non-standard success paths is not taken. However, some applications might be driven by the uncertainty involved in crediting some non-standard success paths.
7. CDF and LERF definitions – the PRA standard allows some flexibility in defining these parameters.	Definitions should be clear and justified. The peer review should be adequate to make sure that reasonable definitions are used. However, non-standard definitions could lead to potential key sources of model uncertainty since it impacts timing and response requirements.
8. Large LOCA long term oxidation in BWRs – since BWRs are designed to maintain 2/3 core height for a very large break LOCA, injection by one LPCI pump into the shroud area may maintain the covered core sub-cooled. Cooling of the top 1/3 core for a substantial time is questionable since long term steam cooling effect may not be ensured.	The body of technical work that supports the assumption of a single LPCI pump for applicable plants should be sufficient on a generic basis for the timeframes considered in a PRA. Additionally alternate success criteria may be employed (e.g., long-term success for LPCI injection also requires implementation of containment flooding strategies as would be required per the SAMGs). However, this could be an application-specific source of uncertainty.
9. Engineering analyses – separate engineering analyses may use codes or invoke other assumptions that may introduce potential sources of modeling uncertainty.	Table A-1 does include the engineering analyses that should be most important (i.e., room heatup and battery depletion calculations). Additionally, unique plant-specific analyses would be expected to be identified as a plant-specific source of model uncertainty. For other engineering analysis, If the codes or methods are accepted by NRC and industry, than the engineering analysis may meet the consensus approach criteria. If not, then the analysis may be a source of modeling uncertainty.
10. Level control during ATWS in BWRs – difficult to perform, but more importantly, the power level achieved in different situations is uncertain. Power/flow oscillations can occur and its impact on the core is uncertain.	The peer review should be adequate to make sure that reasonable success criteria are used. Certainly, there are HRA uncertainties, but those should be able to be subsumed into the general treatment of HEP uncertainties. This could be an application-specific uncertainty, however, especially for an application focused around ATWS mitigation.
11. Post-LOCA boron precipitation in PWRs – modeled in design basis event thermal hydraulic evaluations, but is not always modeled in PRAs.	Research is underway to further clarify this issue. The treatment of this phenomenon will impact long term cooling success criteria following larger LOCAs.

Table A-3 (continued)
Supplementary List for Other Potential Sources of Model Uncertainty

Topic	Discussion of Issue
12. Digital instrumentation and control.	Some plants have incorporated digital systems into their designs or to replace existing analog systems. There are model uncertainties associated with modeling digital systems, such as those related to determining the failure modes of these systems and components.
13. Credit for non-safety related equipment in recovery actions.	This could involve the use of portable equipment and/or the use of flexible hoses. Some of the equipment may be pre-staged and the actions may be procedurally directed, but not all.
14. Passive system degradation mechanisms – aging of active components is incorporated into the periodic data analysis updates but passive system reliability is generally not accounted for.	Plant-specific data based on applicable recent data, representative of plant operation now and in the near future. Otherwise relying on generic data sources. This issue could be important for applications of the model that extrapolate into longer time intervals and pipe failure frequency assessments.
15. Water hammer impacts on system performance.	A water hammer event can cause significant stresses in pipes and components. The analysis of failure of pipes and components given a water hammer event is not generally available. The incorporation or lack of incorporation of the impact of water hammer events may be relevant in some applications of the model.
16. Selection of components in a common cause group.	The choice of common cause groups or lack of identification of some common cause groups could have an influence in certain applications of the model.
17. Capability of battery charger to start and carry loads if the battery is unavailable.	Credit for the charger to start and carry several loads simultaneously may not be appropriate depending on the rating of the charger compared to the rating of the corresponding batteries.
18. Standby failure rate model.	The selection of the standby failure rate model could be relevant for specific applications of the model (specifically for surveillance frequency change evaluations).

A.2.4 Disposition of Sources of Uncertainty

Recall that the list of items that have been earmarked as candidate sources of modeling uncertainty as identified in Tables A-1 through A-3 were differentiated from those items that are related to scope or level of detail that can relate to simplifying assumptions rather than true modeling uncertainty issues. True modeling uncertainty issues would lead to assumptions that are made with the knowledge that a different reasonable alternative assumption exists.

Alternatively, an assumption related to scope or level of detail is simply one that is made for modeling convenience.

Table A-4 provides the list of those items that originally appeared in Appendix H of the Technical Basis Document organized by the corresponding ASME/ANS PRA Standard High Level Requirements. A brief discussion of the issue is provided for each item. The items that have been previously included in Tables A-1 to A-3 as sources of model uncertainty are identified as such. On the other hand, if the item did not appear in Tables A-1 to A-3, then those items have not been identified as sources of modeling uncertainty, and rather are related to scope or level of detail assumptions. Because of their nature, these items will typically not be identified as candidate sources of model uncertainty, but are presented here for completeness compared to the information that originally appeared in the Technical Basis Document.

Table A-4
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
IE-A (Initiating event identification)	***Boron dilution events***	The treatment of boron dilution in PWRs may vary significantly in both the deterministic models and in the quantitative probabilistic model, but is not expected to have a significant impact on the base PRA model. *Boron dilution events have been identified in Table A-3 as a potential source of model uncertainty for some applications.*
	Environmental impacts on initiating events (for example, intake, offsite power, and so on)	Local environment conditions may significantly increase or decrease the frequency of initiators. Proper identification of such environmental impacts is part of the ASME/ANS PRA Standard requirements.
	Grid stability	Additionally, changes in the operation of utility transmission and distribution grids following deregulation may have increased the potential for grid instabilities, which in turn can lead to increases in the likelihood of loss of offsite power events at nuclear generating stations. *Grid stability has been identified in Table A-1 as a candidate source of model uncertainty.*

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
IE-A (Initiating event identification) (continued)	Human-induced initiating events	The crew can induce initiating events that have not been included in the PRA. The crew has access to equipment and instrumentation that can significantly alter the perceived initiating event list. However, the PRA industry has exhaustively identified initiating event categories in countless studies over the past 30 years such that there is a high level of confidence that the relevant initiators have been identified.
	Multi-unit events	Multiple units may provide both significant benefit - by virtue of the sharing of equipment and personnel - and significant challenges if all units require accident mitigation simultaneously. Proper identification of those initiators that impact both units is part of the ASME/ANS PRA Standard requirements.
	Environmental dependencies (ex-plant and in-plant)	The PRA is generally structured to provide an average risk profile of the plant. Initiators may be caused by or adversely impacted by extreme environmental conditions that may not be explicitly accounted for in the base model. Examples include: • Extreme temperatures of air, SW, CST, RWST, and suppression pool (BWR) • Low intake water levels (such as silting) • Zebra mussels or bio-fouling Time frames when extreme conditions exist are addressed as part of the Maintenance Rule implementation at the site.
	Spatial dependencies	Plant walkdowns and design information are used to isolate spatial effects that may induce initiating events.
	Physical dependencies	There may be physical dependencies that are not fully incorporated into the PRA model. However, routine maintenance and updates of the PRA model help to ensure that the models represent the as-built, as-operated plant.

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
IE-A (Initiating event identification) (continued)	***Common cause failures***	Initiators may both induce a plant transient and adversely impact mitigating systems. Common-cause effects may create particularly severe initiating events not typically seen in the operating experience. *Common cause failures have been identified in Table A-1 as a candidate source of model uncertainty in the context of loss of support system initiating events.*
	Initial plant conditions (for example, constant/ changing power level, EOC, BOC, and so on)	This includes consideration of the following: • Power level • Axial power shape • Alignments The base PRA model initiating event identification should not be significantly impacted by these potential variations, however.
	Maintenance/operational activities (for example, switchyard work, system testing)	These activities can present alignments or initiators that are not anticipated in the model. Variations in plant configuration are controlled by the Maintenance Rule, and the plant risk is likely significantly improved.
	Configuration impacts (for example, system alignments, maintenance conditions, FW controller settings, and so on)	These activities can present alignments or initiators that are not anticipated in the model. Unusual alignments imposed by on-line maintenance, testing, or emergent work may create an aggravated initiating event. Variations in plant configuration are controlled by the Maintenance Rule, and the plant risk is likely significantly improved.
	Seasonal impacts (for example, LOOP, loss of SW, and so on)	Specific variations to initiating event frequencies based on seasonal variations are typically not made in the base PRA model, but may be captured if specific seasonal discrepancies are noted in the model development process.
	Changes in plant operational philosophies (for example, more/less on-line maintenance, and so on)	Plant operations and the associated controls could strongly influence the model veracity if substantial changes are made. Routine updates of the PRA models should ensure that the model represents the as-built, as-operated plant.

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(IE-B) Initiating event grouping	Subsumed events	Events that are grouped together and represented by a single initiator that is more limiting than the subsumed events. Initiators must be realistically grouped such that excess conservatism is not included in the grouping of initiating events and that more severe events are not included in a group that does not accurately reflect that more severe conditions of the initiating event. Part of this bounding nature includes accurately characterizing the LERF potential for subsumed events.
	Screened events	Events that are not explicitly modeled in the PRA because they meet the criteria of the ASME/ANS Standard to be eliminated from the quantified model.
	Bounding impacts from grouped events (exclusion of initiators)	Impacts associated with subsumed events that are bounded by assessments for all of the events within the group. This requires considerations related to assumptions regarding what is bounding about an event, for example, a sudden loss of offsite AC versus a slow or intermittent degradation; similarly for loss of air events.
	Types of initiators modeled with thermal hydraulic calculations	Initiating events may introduce a wide spectrum of effects on both the primary and support systems. Plant-specific thermal-hydraulic analysis for specific initiating events is generally limited. In addition, the models may be insufficient to provide the degree of fidelity necessary.
(IE-C) Initiating event frequency estimation	Partial failures	The collection of initiating event data may contain partial failures. The data can be assumed to be a complete failure or can be discarded as not a failure event.
	Applicability of generic data	Generic data can be used to characterize initiating event frequencies or can be used as part of a Bayesian update to incorporate plant-specific experience. In either case, the generic data needs to be appropriate to the specific plant and type of initiating event. *The choice of generic prior data has been included in Table A-3 as a potential source of model uncertainty for some applications.*

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(IE-C) Initiating event frequency estimation (continued)	Applicability of industry experience to environmentally influenced events (that is, loss of service water, LOOP, and so on)	An individual plant may be much less or much more susceptible to environmentally induced initiating events or failures. Examples include: • Plants in typical hurricane zones • Plants in typical tornado zones • Plants using salt water or brackish water for cooling • Plants subjected to heavy snow or ice Meeting the ASME/ANS PRA standard helps to ensure that these plant-specific issues are addressed.
	Applicability of past performance to future operation	Significant changes in any of the following may result in negating the applicability of past performance to future operation: • Plant management • Fuel cycle • Electric power uprate • Climate • Maintenance practices Routine updates of the PRA models should ensure that the model represents the as-built, as-operated plant.

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(IE-C) Initiating event frequency estimation (continued)	***Treatment of rare and extremely rare events***	The ASME/ANS Standard provides rules that allow the screening of rare and extremely rare events. These events are either observed or postulated events that may have significant impacts on the plant and its ability to be safely shut down. In addition, there may be initiators that should be modeled but have not yet been identified as unique or may be special challenges that require separate treatment. The PRA model relies on the quality assurance records of the plant, sound construction practices, and rigorous testing and inspections to ensure that there are no plant flaws that could compromise the mitigation capability of the plant. Therefore, the PRA models do not address this area of completeness uncertainty. On the other hand, there are typically very low likelihood events (e.g. LOCA & ISLOCA) that are included in the PRA models. The frequencies assigned to these items are subject to engineering interpretations of limited data sets. *The LOCA frequency portion of this issue has been identified as a candidate source of model uncertainty in Table A-1. The generic topic for the treatment of rare and extremely rare events issue has been identified in Table A-3 as a potential source of model uncertainty for some applications.*
	Aging	See below.

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(IE-C) Initiating event frequency estimation (continued)	***Active/passive degradation mechanisms***	There can be increases in initiator frequency, severity of challenge, or both. Unusual susceptibility to LOCA may result from the following mechanisms that may not be modeled explicitly nor reflected in past history: • Corrosion • Poor weld repair • Hidden flaws • Aging The effects of these issues are controlled by rigorous testing and inspection programs that are oriented to uncover adverse impacts of plant-age-related phenomena. The PRA relies on these test and inspections to ensure that the as-built plant coincides with the as-designed plant. *However, the issue of passive system degradation mechanisms has been identified in Table A-3 as a potential source of model uncertainty for some applications.*
(AS-A) Accident sequence development	Role of partial failures in accident sequence progression	By incorporating partial failures as complete failures, the accident progression may become conservatively biased. Examples include: partial clogging of strainers or filters; dead-head operation of pumps; pump flow below design. Another aspect of the issue of partial failures (or partial success) is the ability to use equipment, which is generally not modeled, intermittently. This may include: • Intermittent pump operation to minimize room heat-up • Switching a single diesel between two plants or two buses
	As-built plant without major flaws	The PRA model relies on the quality assurance records of the plant, sound construction practices, and inspections to ensure that there are no plant flaws that could compromise the mitigation capability of the plant.

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(AS-A) Accident sequence development (continued)	Thermal hydraulic codes used	The use of an approximate thermal-hydraulic model at a representative time during the fuel cycle is used in the evaluation of system success criteria. The single model is used to represent all plant life. Limitations in the model development include: • Generic versus plant specific • Time in fuel cycle (axial profile and burn-up) • Sophistication of the model • Ability to model different events (for example, large LOCA, ATWS, loss of FW) The ASME/ANS PRA Standard includes requirements to identify thermal hydraulic code limitations to help ensure that they are used appropriately.
	Beyond design basis environment	The analysis of beyond design basis events is much less available than the analysis of design basis events. *The operability of equipment in beyond design basis environments has been identified in Table A-1 as a candidate source of model uncertainty, but under the systems analysis heading rather than accident sequence analysis although it could be applicable to both categories.*
	Long time-frame scenarios (for example, SGTR [PWRs], loss of containment heat removal [BWRs])	PRA modeling techniques are generally tailored to short- or intermediate-term actions. The very long-term actions are not judged to be as well characterized and may be subjected to a substantial increase in uncertainty bands. Two specific issues related to loss of containment heat removal scenarios (i.e., impact of containment venting on systems and success after containment failure) have been separately identified below as candidate sources of model uncertainty.
	Very short time-frame scenarios (for example, ATWS, LBLOCA)	The simulator provides excellent training and feedback to the PRA on very short-term actions.

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(AS-A) Accident sequence development	Performance in beyond design basis conditions	Crew training in the simulator treats a wide variety of beyond design basis conditions. These conditions may not encompass all beyond design basis events. Credit for operator actions under these conditions is established through the use of accepted HRA methodologies per the ASME/ANS PRA Standard.
	Credit for non-procedural recovery actions	Recovery actions can be included in the PRA if justifiable. Justification may include: • Develop a procedure for the recovery action • Training without specific procedure • Skill of the craft
	Impact of LOOP/SBO conditions	Crew response under LOOP and SBO conditions are subjected to a wide variety of influences that may not be easily captured in the PRA, but should be considered as part of the performance shaping factors utilized in the HRA development. These include: • Poor lighting • Poor ventilation • High temperature for personnel • Reduced access • Loss of communication • Significant loss of instrumentation • Loss of computerized systems • Extensive competing actions for restoration of power
	Procedure interpretation	The interpretation of procedures may vary among crews. This can be tested in the PRA HRA development.
	Training	Training sharply affects the crew performance. This results from practice and specific guidance on actions. Whether or not actions are trained should be considered in the HRA development.

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(AS-A) Accident sequence development (continued)	***Moderator temperature coefficient (MTC)***	Changes in plant response may be significantly different than anticipated by the crew because the MTC varies through the fuel cycle. *The MTC treatment issue has been identified in Table A-3 as a potential source of model uncertainty for some applications.*
	Failure criteria (for example, Service Level C) (PWR and BWR)	RPV overpressure failure is not a known failure limit. Service Level C has been used in the past for PRA reasonable estimates of the failure point. *Additionally, however, the issue of the consideration of pressurized thermal shock in PWRs has been identified in Table A-3 as a potential source of model uncertainty for some applications.*
	Effectiveness in adverse environments	Crew ability to effectively think and perform as the environment degrades is difficult to incorporate into the PRA. However, adverse environmental conditions will typically be reflected via the performance shaping factors utilized in the HRA development to attempt to account for this.
	Necessary/available recovery actions	The repair and recovery of failures is an area of significant judgment in the PRA model. It involves the designation of sufficient time, access, personnel, and guidance to either recovery (manual action) or repair of a failed SSC. The accident sequence level of discrimination with regard to plant conditions, timing, operator interface, and use of non-safety systems. The finite nature of the level of delineation collapses the continuum of possible sequences to a limited set. *Note that recovery actions are included in Table A-1 in the context of support system initiating events as a candidate source of model uncertainty.*

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(AS-B) Accident sequence dependencies	***Room heatup calculations***	Room heatup evaluations are subjected to a number of critical variables: • Heat sources (restricted under accident conditions) • Heat sinks (accurate modeling is difficult) • Initial temperatures and outside temperature • Intermittent operation *Room heatup calculations have been identified in Table A-1 as candidate sources of model uncertainty, but under the success criteria heading rather than accident sequence analysis although they could be applicable to both categories.*
	Temperature-dependent failure criteria	The failure of equipment associated with high temperature is subjected to a wide variability. EQ information provides very high confidence that equipment can survive, but survivability above these EQ temperatures is also feasible and realistic. However, this issue is considered to be encompassed within the identification of the room heatup calculations as a candidate source of model uncertainty.
	Battery life	In the context of accident sequence dependencies, this issue leads to assumptions related to the viability of systems to operate without dc power. *The operation of equipment after battery depletion has been identified in Table A-1 as a candidate source of model uncertainty.*
	RCP seal leakage (PWRs)	The RCP seal leakage is a controversial topic for some plants. However, the NRC and industry recently agreed on a consensus model that can be used for representing certain plant types. The RCP seal leakage model affects the RPV inventory capability particularly under SBO conditions. *RCP seal leakage has been identified in Table A-1 as a candidate source of model uncertainty.*

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(AS-B) Accident sequence dependencies (continued)	***Recirculation pump seal leakage (BWRs)***	BWRs that use isolation condensers are sensitive to the recirculation pump seal LOCA size and probability of occurrence. These vary with the type of recirculation pump seals used. *Recirculation pump seal leakage has been identified in Table A-1 as a candidate source of model uncertainty.*
	Accumulator adequacy	Air accumulators are provided for operation of equipment subsequent to the loss of normal pneumatic supplies. The limitations on the accumulator are: • Leakage past valves (for example, check valves) • Number of expected valve cycles Credit for accumulators would be expected to be supported by realistic engineering calculations per the ASME/ANS PRA standard with the results factored into the accident sequence dependency development.
	CST volume	The characterization of the CST or RWST inventory may be either conservative (tech spec requirement) or highly variable. Any need for additional makeup requirements would be expected to be included in the accident sequence dependency development.
	Impact of containment venting on systems (BWRs NPSH)	Adverse impact of containment venting may occur in BWRs due to: • Release of steam to the reactor building • Rapid containment depressurization and loss of NPSH *The impact of containment venting on system NPSH has been identified in Table A-1 as a candidate source of model uncertainty.*
	Multi-unit credit/impact	There may be substantial plant capability that exists within the plant to use ac, dc, or fluid systems via cross-ties. These cross-ties may or may not be procedurally directed and the subject of training exercises. Their use in the PRA should represent a realistic assessment of their likelihood of use.

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(AS-B) Accident sequence dependencies (continued)	Initiator-caused environmental impacts on systems	Loss of HVAC, flooding, and so on can induce personnel error, system failures, or loss of instrumentation. These considerations should be included in the accident sequence development process.
	Time-dependence of failures due to environmental conditions	Time phasing of accident sequences is not generally performed. It may be prudent to perform time-phased sequences for some environmental impacts that are slowly developing. The lack of including these time phased impacts will tend towards a slight conservative bias treatment in the model.
	Recovery of ac power after dc battery depletion	Restoration of ac power generally requires dc power (and pneumatic supplies) in the switchyard. It also requires dc power in the plant for breaker operation. Some manual actions may be sufficiently well trained to be performed event after dc is lost. *The availability of dc power to perform restoration actions has been identified in Table A-1 as a candidate source of model uncertainty (under the Grid Stability category).*

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(AS-B) Accident sequence dependencies (continued)	Functional dependencies	The identification of system dependencies based on the sequence of events is typically included in the accident sequence development process. Examples include the following: • Water hammer in discharge line • RPV overfill (induced LOCA or induced failures) • Room-cooling loss cause high temperature isolation • Steam tunnel temperature causes high temperature isolation • Pump operation on minimum flow discharges CST (RWST) volume (discharge of CST to suppression pool [sump] through minimum flow valve) • Low volume system adequacy for RPV level control • Operation of RHR in suppression pool cooling given high drywell pressure or reactor low level • Timing of bleed and feed operation

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(AS-B) Accident sequence dependencies (continued)	Sequence dependencies causing system effects	The accident sequence defines the environment, the system functional failures, timing, and the crew impacts. These, in turn, affect the systems. Sequence dependencies causing system effects may include the following and as such need to be considered in the accident sequence development process. • Direct failure • Increased number of demands • Auto alignment • Loss of auto alignment • Automatic realignment • Failure modes – Water hammer – Steam binding – Air binding – Accumulator depletion
	Time-dependent success criteria (for example, time phasing)	Accident sequences may have different systems available or recoveries introduced during the time of the accident progression. The ability or lack of ability, to model the sequence of time phases is referred to as *time phasing*. Examples of time phasing issues that may or may not be incorporated in the accident sequence development are: • Assuming that a “run” failure always occurs at t = 0 • Credit for additional personnel on-site, for example, TSC

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(SC-A) Overall success criteria	***Credit for nonstandard success paths (for example, alternative makeup sources)***	The PRA takes credit for safety and non-safety systems for accident mitigation. In some cases, alternative systems that are infrequently or never tested for the application are credited. The alternative system may have limitations such as discharge head that limit their range of usefulness. On the other hand, alternative systems may be given too little credit by assigning high crew failure rates to align. *Credit for non-standard success paths has been identified in Table A-3 as a potential source of model uncertainty for some applications.*
	Success following containment failure (BWR)	Following the analysis developed in NUREG-1150 (NUREG/CR-4550), a probabilistic model provides the probability of continued system operation after containment leaks or rupture. *Success following containment failure has been identified in Table A-1 as a candidate source of model uncertainty.*
	Definition of core damage	The definition of what constitutes a core damage end state is critical to the effective communication of the Level 1 PRA results. Examples of areas of potential disagreement include: • LBLOCA reflood (PWRs) • LBLOCA long-term oxidation (BWRs) • Power/flow oscillation (ATWS) *The definition of core damage has been identified in Table A-3 as a potential source of model uncertainty for some applications.* *Additionally, the specific issue related to the Large LOCA long term oxidation in BWRs has also been identified in Table A-3 as a potential source of model uncertainty for some applications.*

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(SC-B) Thermal/hydraulic, structural, and other supporting engineering bases	T/H model (core, RCS, containment, and so on)	The use of an approximate thermal hydraulic model at a representative time during the fuel cycle is used in the evaluation of system success criteria. The single model is used to represent all plant life. Limitations in the model development may include: • Generic versus plant-specific • Time in fuel cycle (axial profile and burn-up) • Sophistication of the model (nodalization) • Ability to model different events (for example, large LOCA, ATWS, loss of FW) • Ability to calculate containment pressures and temperatures, for example, dependency on heat sink and heat source models. The ASME/ANS PRA Standard includes requirements to identify thermal hydraulic code limitations to help ensure that they are used appropriately.
	Use of generic T/H models	Thermal-hydraulic analysis is generally performed with plant-specific models. There may be cases where generic information is used to supplement this plant-specific modeling. ATWS modeling is one example.
	Use of generic containment structural analyses	The containment structural analysis performed generically is sometimes used to represent a plant-specific situation. This may apply to all of the required analyses or only to a portion of the analysis, such as dynamic loading.
	Credit for repair and recovery	Repair and recovery of failures is an area of significant judgment in the PRA model. It involves the designation of sufficient time, access, personnel, and guidance to either recovery (manual action) or repair of a failed SSC. The ASME/ANS PRA Standard has specific requirements regarding taking credit for repair and recovery.

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(SC-B) Thermal/hydraulic, structural, and other supporting engineering bases (continued)	Plant-specific characteristics (for example, pump curves, axial power shape, burn up)	Plant-specific features are incorporated in the PRA model and the thermal-hydraulic analysis. However, there is always a selection made regarding the time during the fuel cycle to be represented. Care must be taken to ensure that these selections do not adversely skew the success criteria development.
	Thermal hydraulic codes	The ASME/ANS PRA Standard includes requirements to identify thermal hydraulic code limitations to help ensure that they are used appropriately. *In any event, separate engineering analysis are identified in Table A-3 as a potential source of model uncertainty for some applications.*
	Structural analyses	Plant-specific analysis adds significant confidence to the assessment but may still be limited in the failure modes considered.
	Room heat-up calculations	Room heatup evaluations are subjected to a number of critical variables: • Heat sources (restricted under accident conditions) • Heat sinks (accurate modeling is difficult) • Initial temperatures and outside temperature • Intermittent operation *Room heatup calculations have been identified in Table A-1 as a candidate source of model uncertainty.*
	Battery life calculations	The duration of a battery to support plant response without the charger is subjected to considerable judgment. This duration is a function of: 1) the load profile applied to the battery, and 2) the battery's initial condition. *Battery life calculations have been identified in Table A-1 as a candidate source of model uncertainty.*

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(SC-B) Thermal/hydraulic, structural, and other supporting engineering bases (continued)	***Number of PORVs required for bleed and feed (PWRs)***	Thermal-hydraulic calculations and the associated models may lead to different success criteria regarding the number of PORVs that must be available under the worst-case circumstances to support bleed and feed. *The number of PORVs required for bleed and feed has been identified in Table A-1 as a candidate source of model uncertainty.*
	Reliance on design basis calculations	The use of design-basis calculations to support success criteria and accident sequence timing may introduce a bias into the PRA calculation. This bias may be reflected in more limiting success criteria and response times than what might be justified using realistic models.
	Cross-ties	There may be substantial plant capability that exists within the plant to use ac, dc, or fluid systems via cross-ties. These cross-ties may or may not be procedurally directed and the subject of training exercises. Their use in the PRA should represent a realistic assessment of their likelihood of use.
	Containment sump/strainer performance	Strainer clogging is a critical failure mode because it may cause failure of redundant equipment. The clogging mechanisms are not completely predictable for beyond design basis events. *Containment sump/strainer performance has been identified in Table A-1 as a candidate source of model uncertainty.*
	Impact of failure of pressure relief	Overpressure of the RPV may result in rupture or leak or pressure relief via the head seal. • Failure criteria (for example, Service Level C) (PWR and BWR) *The impact of pressure relief has been identified in Table A-1 as a candidate source of model uncertainty.*

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(SC-B) Thermal/hydraulic, structural, and other supporting engineering bases (continued)	***Impact of containment venting on systems (BWRs NPSH)***	Adverse impact of containment venting may occur in BWRs due to: • Release of steam to the reactor building • Rapid containment depressurization and loss of NPSH *The impact of containment venting on systems has been identified in Table A-1 as a candidate source of model uncertainty.*
	ATWS modeling	Plant-specific detailed ATWS models are generally not available to support realistic success criteria. • Power level versus water level, ECCS overfill, boron mixing (BWR) • Moderator temperature coefficient (MTC) *Modeling of level control during ATWS conditions in BWRs has been identified in Table A-3 as a potential source of model uncertainty for some applications.*
	Post-LOCA boron precipitation (PWRs)	The ability of PWRs to assure safe shutdown given the failure mode of Boron precipitation may not always be addressed in the PRA. *Post-LOCA boron precipitation in PWRs has been identified in Table A-3 as a potential source of model uncertainty for some applications.*
	Initial power level	Lower power levels introduce the following: • Lower decay heat levels • Different initial plant alignments than modeled in the full power configuration (such as FW pumps aligned, condenser, and recirculation pumps operating) The typical PRA approach for the base PRA model is to assume that the sequence of events starts from 100% power levels.
	Time in core life	Burnup and axial power shape may influence decay heat, time-to-core damage, and fission product inventory. Care must be taken to ensure that these selections do not adversely skew the success criteria development.

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(SC-B) Thermal/hydraulic, structural, and other supporting engineering bases (continued)	Set point drift	Set-point drift impact on premature or delayed system operation is not generally included.
	Time of year	The time during the year may influence the configuration-specific risk profile. Examples include: • Weather effects • Grid stability • Evacuation time Specific variations in the system modeling due to seasonal variations are typically not made in the base PRA model, but may be captured if specific seasonal discrepancies are noted in the system model development process.
(SY-A) System failure modes and failure causes	Super components	The modeling of systems using groups of components. This is generally done at the level for which data is available. For example, the diesel generator can be modeled as a single component. This also can refer to the grouping of a number of disparate components into a group and modeled as a single contribution to a system failure. Data to characterize the super component must be developed consistent with the super component boundary definition.
	Use of generic "black box" models (for example, RPS, rod insertion, and so on)	Some systems are sufficiently complex that the system itself is represented by a qualitative estimate of its reliability rather than a detailed model. The system that generally falls into this category is the reactor protection system (RPS).
	Treatment of equipment repair	Equipment repair may be incorporated into the model using plant-specific or generic data. The ASME/ANS PRA Standard recognizes this as a viable option. Because of the lack of data, however, there may be conservative biases introduced into the model.
	Credit for manual operation or local operation (valves, breakers)	Certain local manual operations are appropriate to credit within the context of the HRA. Nevertheless, the variations in access, timing, guidance, and available trained personnel may limit this effectiveness. These considerations should be factored into the HRA analysis.

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(SY-A) System failure modes and failure causes (continued)	Component boundaries	The development of data requires a clear and precise definition of the component boundary. The available data may not, in some cases, have the boundary readily defined. In these cases, the data selection should be made to ensure that the choice does not adversely skew the results.
	Modeling of unique components (data applicability)	Unique plant components may not have adequate data to represent them in the quantitative model. Again, the data selection should be made to ensure that the choice does not adversely skew the results. *However, the modeling of digital instrumentation and control has been identified in Table A-3 as a potential source of model uncertainty for some applications.*
	Use of spare equipment	Spare batteries, chargers, and so on represent a real beneficial plant aspect that can be included in the PRA model to promote realism. The alignment of this equipment should address the access, timing, training, guidance, and available personnel to complete the alignment. This is also related to sequence time phasing.
	Design or construction flaws	The PRA model relies on the quality assurance records of the plant, sound construction practices, and inspections to ensure that there are no plant flaws that could compromise the mitigation capability of the plant; therefore, system, structures, components, and initiating events do not address this area of nodal uncertainty. Design or construction flaws are controlled by rigorous testing and inspection programs that are oriented to uncover these “flaws” before they impact safety. The PRA relies on these tests and inspections to ensure that the as-built plant coincides with the as-designed plant.
	System capabilities (flows, capacities, and so on)	System flows and tank capacities are generally treated as realistically as possible.

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(SY-A) System failure modes and failure causes (continued)	Flow diversions	Flow diversion can create a situation where a system is not capable of supplying the minimum flows required. The approach to flow diversion includes: • Use of realistic flow requirements • Allowance for some flow diversion without materially affecting the ability to supply sufficient flow • Consideration of the high-pressure to low-pressure flow diversion as a special case that needs a separate calculation
	Treatment of instrumentation required for operator actions	The crew's window on the plant comes primarily from instrumentation. Failures of instrument or degraded conditions of instrumentation may significantly alter the way the crew responds to an accident, but the level of redundancy in the instrumentation should be considered as part of the performance shaping factors utilized in the HRA development.
	Alternative systems	The PRA takes credit for safety and non-safety systems for accident mitigation. In some cases, alternative systems that are moved or infrequently tested for the application are credited. In general, the peer review should be adequate to ensure that unreasonable credit for alternative systems is not taken. *However, credit for non-safety related equipment in recovery actions has been included in Table A-3 as a potential source of model uncertainty for some applications.*
	Offsite resources	The realistic incorporation of offsite resources into the HRA is complicated by the following: • Sequence time phasing • Variability from plant to plant • Impact of weather • Impact of time of day *The availability of offsite resources is captured via the identification of credit for the emergency response organization in Table A-1 as a candidate source of model uncertainty.*

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(SY-A) System failure modes and failure causes (continued)	Alignments	Not all possible alignments are included in the model. Variations in plant configuration are controlled by the Maintenance Rule, and the plant risk is likely significantly improved.
	Active/passive failure mechanisms	See discussion below for active/passive degradation mechanisms.
	Active/passive degradation mechanisms	Component degradations that lead to failures that are the result of corrosion, poor inspections, hidden flaws, aging, safety culture, and so on are not reflected in past history and can significantly impact associated uncertainty on the reliability values. *The issue of passive system degradation mechanisms has been identified in Table A-3 as a potential source of model uncertainty for some applications.*
	Dynamic system response modeling limited by Boolean logic models	Boolean logic models are static models that reflect a specific configuration or average state. This limits the ability to model changes in plant configurations as a function of time during the event. This typically results in slight conservative bias treatments.
(SY-B) Common-cause failures and intersystem and intra-system dependencies	***Operability of equipment in beyond design basis environments***	The use of design basis calculations to support success criteria and accident sequence timing may introduce a bias into the PRA calculation. This bias is reflected in more limiting success criteria and response times than can be justified using realistic models. *The operability of equipment in beyond design basis environments has been identified in Table A-1 as a candidate source of model uncertainty.*
	Operability of equipment given a loss of room cooling	The failure of equipment associated with high temperature is subjected to a wide variability. EQ information provides very high confidence that equipment can survive, but survivability above these EQ temperatures is also feasible and realistic. *The operability of equipment given a loss of room cooling has been identified in Table A-1 as a candidate source of model uncertainty (under the operability of equipment in beyond design basis environments category).*

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(SY-B) Common-cause failures and intersystem and intra-system dependencies (continued)	***Operability of equipment given loss of component cooling (CCW, SW, and so on)***	Component cooling can be necessary for equipment operation. However, the timing of when equipment might fail given the loss of cooling is not generally available. This is an area where time phasing of the accident sequence could increase realism. *The operability of equipment given a loss of component cooling has been identified in Table A-1 as a candidate source of model uncertainty (under the operability of equipment in beyond design basis environments category).*
	Operation of pumps without flow (for example, deadheading of low-pressure pumps in small LOCAs)	The need for minimum flow line operation is a controversial modeling question. Empirical evidence seems to indicate that pumps can operate deadheaded for extended times. This may be very pump specific. *The operation of pumps without flow has been identified in Table A-1 as a candidate source of model uncertainty (under the operability of equipment in beyond design basis environments category).*
	Water hammer impacts on system performance	One of the dynamic loads that is postulated to occur in a nuclear power plant is the water hammer event. The water hammer event can cause significant stresses in pipes and components. The analysis of failure of pipes and components given a water hammer event is not generally available. *Water hammer impacts on system performance have been identified in Table A-3 as a potential source of model uncertainty for some applications.*
	Multi-unit interactions	Multiple units may provide both significant benefit by virtue of the sharing of equipment and personnel and significant challenges if all units require accident mitigation simultaneously. There may be substantial plant capability that exists within the plant to use ac, dc, or fluid systems via cross-ties. These cross-ties may or may not be procedurally directed and the subject of training exercises. There use in the PRA should represent a realistic assessment of their likelihood of use.

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(SY-B) Common-cause failures and intersystem and intra-system dependencies (continued)	***Common cause failure groups (intra-system, inter-system)***	Common cause failures can be important contributors to the PRA. Inter-system common cause failures are generally not included and are only required to meet Category III of the ASME/ANS PRA Standard. However, the Standard does include requirements to ensure that the CCF groups are chosen appropriately. *In any event, the selection of components in a common cause group has been included in Table A-3 as a potential source of model uncertainty for some applications.*
	Time-dependence of system failures due to system interdependencies or environmental conditions	Accident sequences may have different systems available or recoveries introduced during the time of the accident progression. The ability or lack of ability, to model the sequence of time phases is referred to as time phasing. Examples of time phasing issues that may or may not be incorporated in the system modeling are: • Diesel generator recovery • Restoration of equipment following initiator or system failure (for example, air, power, ac bus, dc bus)
	DC power dependence on chargers and batteries	A difficult success criterion to establish is the need for both batteries and chargers for dc power continuity. There are plants where the batteries are not required as long as the chargers are available. *The capability of battery charger to start and carry loads if the battery is unavailable has been included in Table A-3 as a potential source of model uncertainty for some applications.*
	Subtle interactions (NUREG/CR-4550 Vol. 1)	There are sneak circuits and dependencies that are difficult to uncover based solely on design basis documents. These subtle interactions may introduce uncertainties into the model. However, thorough system modeling per the ASME/ANS PRA Standard helps to ensure that all relevant dependencies are included in the system models.

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(HR-A) Pre-initiator identification ***Note that the basis for the HEP values as listed below is captured in Table A-2 as a general source of model uncertainty. This treatment is chosen in lieu of identifying each of the specific topics in this category individually.***	HFE delineation	The discrimination of those HFEs that are to be modeled and the conditions under which they are characterized. There are hundreds of individual HFEs that could be modeled. Of these, there are HFEs that are screened or subsumed into larger groups. The larger group of HFEs is then typically represented by a single set of limiting conditions.
	HFE applicability	The HFE application to specific circumstances within the accident sequence may be constrained in different ways for different applications.
	Crew-to-crew variability	Crew-to-crew variability is generally not included as part of the HRA. • Overall experience • Experience with event(s) • Staffing level (minimum versus maximum) • Back shift maintenance resources • Crew personalities • Creativity
	Organizational interfaces	The plant-specific organization during an event may be difficult to capture in the HRA and may strongly depend on the personalities involved, including: • Operations-Maintenance • Staff-Management • Control Room-TSC • Ex-Plant (for example, grid operator)
	Errors of commission	Errors of commissions are not explicitly included (with some exceptions). Errors of commission can vary widely and result in extreme conditions in the plant.
	Procedural changes (permanent and temporary)	Temporary procedures and alignments, night orders, and so on are not generally accounted for.

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(HR-B) Pre-initiator screening ***Note that the basis for the HEP values as listed below is captured in Table A-2 as a general source of model uncertainty. This treatment is chosen in lieu of identifying each of the specific topics in this category individually.***	Worker-machine interface	There may be unique components, instruments, or controls that make plant operation, accident response, and recoveries significantly better or worse than the typical plant. These shaping factors are difficult to fully integrate into the HRA.
	Training and procedures	Training and procedures form the basis for the HRA.
	Multi-unit events	Multiple units may provide both significant benefit—by virtue of the sharing of equipment and personnel—and significant challenges if all units require accident mitigation simultaneously.
	Crew response times	The simulator, crew input, and JPM response times are sources of information for crew response times. All sources are not consistent and can be either optimistic or pessimistic.
	Distractions (for example, tired, problems outside of work, and so on)	The crew work schedule and individual crew member conditions are not generally included as part of the shaping factors of the HRA.
	Crew turnover	Period of crew turnover and the information transmittal at crew turnover is not modeled.
	Crew awareness to conditions	Training can alter crew awareness. The awareness of the crew to specific accident conditions varies with the training cycle and current industry experiences that are promulgated to the crews.
	Circadian clock	Time of day is not generally included in the HRA despite evidence that the most serious crew errors occur between 12 midnight and 6 a.m.
	Training cycle emphasis	Training can alter crew awareness. The awareness of the crew to specific accident conditions varies with the training cycle and current industry experiences that are promulgated to the crews.
(HR-C) Pre-initiator characterization	--	--

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(HR-D) Pre-initiator quantification	***Basis for pre-initiator HEPs***	Pre-initiator human error probabilities (HEPs) depend on a generic methodology (for example, NUREG/CR-1278 or ASEP). The method, while leading to consistent approaches and quantification, may not adequately address plant-specific variables or changing maintenance practices. *Note that as indicated above, the basis for the HEP values is captured in Table A-2 as a general source of model uncertainty. This treatment is in lieu of identifying each of the specific topics in this category individually.*
(HR-E) Post-initiator identification ***Note that the basis for the HEP values as listed below is captured in Table A-2 as a general source of model uncertainty. This treatment is chosen in lieu of identifying each of the specific topics in this category individually.***	HFE delineation	The discrimination of those HFEs that are to be modeled and the conditions under which they are characterized. There are hundreds of individual HFEs that could be modeled. Of these, there are HFEs that are screened or subsumed into larger groups. The larger group of HFEs is then represented by a single set of limiting conditions.
	HFE applicability	The HFE application to specific circumstances within the accident sequence may be constrained in different ways for different applications.
	Scenario-dependent recovery and repair	The accident sequence level of discrimination with regard to plant conditions, timing, operator interface, and use of non-safety systems. The finite nature of the level of delineation collapses the continuum of possible sequences to a limited set. Repair and recovery of failures is an area of significant judgment in the PRA model. It involves the designation of sufficient time, access, personnel, and guidance to either recovery (manual action) or repair of a failed SSC.

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(HR-E) Post-initiator identification (continued)	Crew-to-crew variability	Crew-to-crew variability is generally not included as part of the HRA. • Overall experience • Experience with event(s) • Staffing level (minimum versus maximum) • Back shift maintenance resources • Crew personalities • Creativity
	Organizational interfaces	The plant-specific organization during an event may be difficult to capture in the HRA and may strongly depend on the personalities involved, including: • Operations-Maintenance • Staff-Management • Control Room-TSC • Ex-Plant (for example, grid operator)
	Errors of commission	Errors of commissions are not explicitly included (with some exceptions). Errors of commission can vary widely and result in extreme conditions in the plant.
	Procedural changes (permanent and temporary)	Temporary procedures and alignments, night orders, and so on are not generally accounted for.
(HR-F) Post-initiator characterization ***Note that the basis for the HEP values as listed below is captured in Table A-2 as a general source of model uncertainty. This treatment is chosen in lieu of identifying each of the specific topics in this category individually.***	Human performance impact of beyond design basis conditions and environments (for example, SGTR, SBO, and ATWS)	The characterization of human performance for beyond design basis events is critical to the successful realism in a PRA. The simulator training and results from that training can support the HEP characterization.

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(HR-F) Post-initiator characterization (continued)	Instrumentation response resulting in degraded information flow to crew	The crew's window on the plant comes primarily from instrumentation. Failures of instrument or degraded conditions of instrumentation may significantly alter the way the crew responds to an accident, but the level of redundancy in the instrumentation should be considered as part of the performance shaping factors utilized in the HRA development.
	Worker-machine interface	There may be unique components, instruments, or controls that make plant operation, accident response, and recoveries significantly better or worse than the typical plant. These shaping factors are difficult to fully integrate into the HRA.
	Training and procedures	Training and procedures form the basis for the HRA.
	Multi-unit events	Multiple units may provide both significant benefit—by virtue of the sharing of equipment and personnel—and significant challenges if all units require accident mitigation simultaneously.
	Crew response times	The simulator, crew input, and JPM response times are sources of information for crew response times. All sources are not consistent and can be either optimistic or pessimistic.
	Distractions (for example, tired, problems outside of work, and so on)	The crew work schedule and individual crew member conditions are not generally included as part of the shaping factors of the HRA.
	Crew turnover	Period of crew turnover and the information transmittal at crew turnover is not modeled.
	Crew awareness to conditions	Training can alter crew awareness. The awareness of the crew to specific accident conditions varies with the training cycle and current industry experiences that are promulgated to the crews.
	Circadian clock	Time of day is not generally included in the HRA despite evidence that the most serious crew errors occur between 12 midnight and 6 a.m.
	Training cycle emphasis	Training can alter crew awareness. The awareness of the crew to specific accident conditions varies with the training cycle and current industry experiences that are promulgated to the crews.

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(HR-G) Post-initiator quantification	***Basis for post-initiator HEPs***	Post-initiator HEPs depend on training practices or emphasis. The method, while leading to consistent approaches and quantification, may not adequately address plant-specific variables or changing maintenance practices. *Note that as indicated above, the basis for the HEP values is captured in Table A-2 as a general source of model uncertainty. This treatment is in lieu of identifying each of the specific topics in this category individually.*
	Intra-crew dependence	The dominant influence of a single individual within the crew may adversely impact recovery if misdiagnosis has occurred.
	Inter-crew dependence	Generally no inter-crew dependence is accounted for.
	Inter-HFE dependence	The ability to evaluate and systematically quantify HFE dependence is more of an art than a science. The existing guidance, while considered to constitute a consensus model, is also considered a source of uncertainty.
	Recovery and repair dependence on specific scenario	The repair and recovery of failures is an area of significant judgment in the PRA model. It involves the designation of sufficient time, access, personnel, and guidance to either recovery (manual action) or repair of a failed SSC. The accident sequence level of discrimination with regard to plant conditions, timing, operator interface, and use of non-safety systems can significantly impact associated uncertainty. The finite nature of the level of delineation collapses the continuum of possible sequences to a limited set. Repair and recovery of failures is an area of significant judgment in the PRA model. It involves the designation of sufficient time, access, personnel, and guidance to either recovery (manual action) or repair of a failed SSC.
(HR-H) Recovery actions	Basis for recovery probabilities	Repair and recovery of failures is an area of significant judgment in the PRA model. It involves the designation of sufficient time, access, personnel, and guidance to either recovery (manual action) or repair of a failed SSC. Availability of appropriate specialty personnel to perform certain recovery or repair actions during graveyard shift or holidays may not be modeled.

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(HR-H) Recovery actions (continued)	Basis for repair probabilities	See discussion above for the basis for recovery probabilities.
(DA-A) Data parameter definition	Super components	The modeling of systems using groups of components. This is generally done at the level for which data is available. For example, the diesel generator can be modeled as a single component. This also can refer to the grouping of a number of disparate components into a group and modeled as a single contribution to a system failure. Data to characterize the super component must be developed consistent with the super component boundary definition.
	Component boundary	The development of data requires a clear and precise definition of the component boundary. The available data may in some cases not have the boundary readily defined. In these cases, the data selection should be made to ensure that the choice does not adversely skew the results.
	Constant failure rate model	The PRA generally includes the assumption that component failure rates are constant (that is, unaffected by plant age, time in fuel cycle).
	Standby failure rate model versus demand failure rate model	The derivation of component failure probabilities in PRAs may use any of the following: • Demand failures • Standby failure rate • Combination of the two to represent the two types of stresses on the component *The standby failure rate model has been included in Table A-3 as a potential source of model uncertainty for some applications.*
(DA-B) Data grouping	Applicability of component type data	Component data may use a type code that combines a broad spectrum of similar components within a single group for both data evaluation and application to a specific plant's SSCs.
(DA-C) Data collection and selection	Partial failures	The use of data that contain partial failures. The data can be assumed to be a complete failure or can be discarded as not a failure event.

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(DA-C) Data collection and selection (continued)	***Applicability of generic data***	Generic data can be used to characterize component failures or can be used as part of a Bayesian update to incorporate plant-specific experience. In either case, the generic data needs to be appropriate to the specific plant and component. Reasons for non-applicability could include: • Components are a significantly different design. • Environmental conditions are significantly different. • Generic maintenance terms represent average values and generally do not reflect a specific plant's maintenance practices or resultant value. *The choice of generic prior data has been included in Table A-3 as a potential source of model uncertainty for some applications.*
	Applicability of past performance to future operation (rectification)	Significant changes in any of the following may result in negating the applicability of past performance to future operation: • Plant management • Fuel cycle • EPU • Climate • Maintenance practices Rectification of past failures usually results in a substantial decrease in the failure probability or complete elimination of a failure mode. Successful rectification of past failure modes is sometimes difficult to document.

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(DA-C) Data collection and selection (continued)	Aging	The service exposure of systems, structures, and components can result in changes in their failure rate. This is represented conceptually by the typical bathtub curve that is commonly used to represent the life response of a component. Early in life, the failure rate may decrease; and late in life, the failure rate may increase. The effects of plant aging are controlled by rigorous testing and inspection programs that are oriented to uncover adverse impacts of plant-age-related phenomena. The PRA relies on these test and inspections to ensure that the as-built plant coincides with the as-designed plant.
	Active/passive degradation mechanisms	Component degradation that leads to failures that are the result of corrosion, poor inspections, hidden flaws, aging, safety culture, and so on are not reflected in past history and can significantly impact associated uncertainty on the reliability values. *The issue of passive system degradation mechanisms has been identified in Table A-3 as a potential source of model uncertainty for some applications.*
	Repair data	Repair and recovery of failures is an area of significant judgment in the PRA model. It involves the designation of sufficient time, access, personnel, and guidance to either recovery (manual action) or repair of a failed SSC. The ASME/ANS PRA Standard has specific requirements regarding taking credit for repair and recovery.
	Recovery data	See discussion above for repair data.
	Time since last test	The average PRA model does not reflect specific times during test cycle.
(DA-D) Data estimation	Treatment of highly reliable components	Some components may be considered so reliable or redundant that their failures have been truncated from the model.

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(DA-D) Data estimation (continued)	Common-cause failure event screening	The screening of observed common-cause events from application to a specific plant is subject to considerable judgment. It involves the elimination of observed events without consideration of the new or different events that could result for the specific plant under evaluation. This is not identified as a unique source of model uncertainty since intra-system CCF events are included as a general source of model uncertainty below.
	Intra-system common cause events	Significant impact on quantified model is anticipated due to common-cause failures within a redundant system. *Intra-system common cause events have been identified in Table A-2 as a general source of model uncertainty.*
	Inter-system common-cause events	Inter-system CCF not generally included. It is only required for Capability Category III of the ASME/ANS PRA Standard.
	Degraded equipment conditions (that is, component known to have performance issues not reflected in data)	Degraded performance of BOP and standby systems may be tolerated for short periods of time. These conditions are generally not reflected in the base PRA model.
(IF-A) Flood area identification	Flood area definition	The zones, components, or area that is used to evaluate the impact of flood. This usually corresponds to an area defined by specific physical boundaries that would contain the flood, but could also be extended to incorporate areas that would be affected by flood propagation.
(IF-B) Flood source identification	Floor drain impacts	Floor drains have two impacts: 1) they remove fluid accumulation, and 2) they provide a flood propagation path via back flow to other compartments.
(IF-C) Flood scenario development	Screened events	Floods that are not explicitly quantified in the model because they meet the test of the ASME/ANS PRA Standard to be eliminated from further consideration.

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(IF-C) Flood scenario development (continued)	Subsumed events	Initiators must be realistically grouped such that excess conservatism is not included in the grouping of initiating events and that more severe events are not included in a group that does not accurately reflect that more severe conditions of the initiating event.
	Flood scenario characterization	The flood scenario characterization is the culmination of all of the considerations listed below such as mitigation, propagation pathways, spray protection, procedures, training, flood flow rates, and so on.
	Flood mitigation	The systems utilized and operator actions along with the associated response time for mitigation of floods should be considered in the internal flooding analysis per the requirements in the ASME/ANS PRA standard.
	Pathways	Identification of flood propagation pathways depends on the use of design information and walkdowns. The AMSE/ANS PRA Standard has specific requirements to consider flood propagation pathways in the internal flooding model development process.
	Spray protection	The source of water spray, its characteristics, and the protection available for electrical equipment to avoid spray-induced failures is generally a matter of using standard rules and judgment. This can result in optimistic or pessimistic characterizations.
	Procedures	The flood response procedures should provide a plant-specific characterization of flood mitigation. The implementation of the procedures depends on recognition of the flood, access, personnel availability, and training. These considerations should be factored into the HEP development for the performance of the flood response actions.
	Training	The crew response to floods may be strongly influenced by the degree of training incorporated into the curriculum. Consideration of the level of training should be factored into the HEP development for the performance of the flood response actions.

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(IF-C) Flood scenario development (continued)	Structural analysis of doors and flood barriers	Flood barriers generally do not have significant calculations to support their integrity under severe flood events. Generic assumptions regarding the expected behavior of doors and barriers may be utilized in the scenario development process. Refined analysis may be warranted in some cases.
	Propagation pathways	Identification of flood propagation pathways depends on the use of design information and walkdowns. The AMSE/ANS PRA Standard has specific requirements to consider flood propagation pathways in the internal flooding model development process.
	Flood flow rates	Flood size or flow rate is characterized by a limited set of data and the judgment of how to use that data. *The flood flow rate issue is considered to be encompassed within the flood size distribution issue (see below) which has been included in Table A-1 as a candidate source of model uncertainty under the broader topic of piping failure mode.*
	Flood accumulation rates	Flood accumulation depends on assumed flow rates and propagation paths out of the zone.
	Spray impacts	The source of water spray, its characteristics, and the protection available for electrical equipment to avoid spray-induced failures is generally a matter of using standard rules and judgment. This can result in optimistic or pessimistic characterizations.
	Water hammer leading to flood	One of the dynamic loads that is postulated to occur in a nuclear power plant is the water hammer event. The water hammer event can cause significant stresses in pipes and components. The analysis of failure of pipes and components given a water hammer event is not generally available. However, there are typically water hammer events included in the generic data employed in internal flooding analysis.
	Detection and diagnosis	Instrumentation accuracy and availability is generally assumed.

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(IF-C) Flood scenario development (continued)	Isolation of flood	The human reliability analysis of flood isolation depends on the ability to identify the event, the source, and a way to isolate it. These are difficult impacts to realistically obtain over the spectrum of floods. Bounding analyses may suffice in some cases which may lead to a slight conservative bias treatment.
	Effectiveness in adverse environments	Crew response under adverse environments is usually conservatively treated when the environment can be identified.
	Flood propagation pathways (such as floor drain backup)	Identification of flood propagation pathways depends on the use of design information and walkdowns. The AMSE/ANS PRA Standard has specific requirements to consider flood propagation pathways in the internal flooding model development process.
	Barrier failure/unavailability	Generic assumptions regarding the expected behavior of doors and barriers may be utilized in the scenario development process. Refined analysis may be warranted in some cases. The AMSE/ANS PRA standard with the RG-1.200 clarifications also necessitates looking at the impacts of potential maintenance alignments that may cause barrier unavailability.
	Multi-unit impacts	Consideration of events that could impact both units simultaneously should be made as part of the flood scenario development process.
	Configuration impacts (for example, maintenance alignments, maintenance conditions)	Not all possible alignments are included in the model. However, the AMSE/ANS PRA standard with the RG-1.200 clarifications does necessitate looking at the impacts of potential maintenance alignments that may cause barrier unavailability.
(IF-D) Flooding-induced initiating events identification and estimation	***Flood frequency data***	Internal floods are relatively rare events with limited applicable data to characterize both the initiator and the flow rate of the event. *Flood frequency data has been included in Table A-1 as a candidate source of model uncertainty under the broader topic of piping failure mode.*

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(IF-D) Flooding-induced initiating events identification and estimation (continued)	***Flood size distribution***	Flood size or flow rate is characterized by a limited set of data and the judgment of how to use that data. *Flood size distribution has been included in Table A-1 as a candidate source of model uncertainty under the broader topic of piping failure mode.*
	Piping failure mechanisms	The incorporation in the model of all the appropriate plant-specific pipe failure mechanisms may result in significant insights regarding plant risk spectrum. *Piping failure mechanisms have been included in Table A-1 as a candidate source of model uncertainty under the broader topic of piping failure mode.*
	Active/passive failure mechanisms	Initiators that are the result of corrosion, poor inspections, hidden flaws, aging, safety culture, and so on are not reflected in past history. These can be increases in frequency, severity of challenge, or both. This source of uncertainty with respect to internal flooding is considered to be encompassed within the piping failure mode category above.
	Aging	The service exposure of systems, structures, and components can result in changes in their failure rate. This is represented conceptually by the typical bathtub curve that is commonly used to represent the life response of a component. Early in life, the failure rate may decrease; and late in life, the failure rate may increase. This source of uncertainty with respect to internal flooding is considered to be encompassed within the piping failure mode category above.
	Inspection frequency and type	The average PRA model does not reflect specific times during test cycle.
(IF-E) Flood-induced accident sequences quantification	--	--
(QU-A) Quantification of core damage frequency	--	--

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(QU-B) Quantification models and codes	Truncated sequences/cutsets	The quantification of the PRA model may have limitations with regard to how many sequences or cutsets are to be retained in the model. The elimination of sequences or cutsets results in the loss of information.
	Rare event approximation	The assumption used in Boolean logic computer codes that the probabilities of failure events are very low and therefore certain approximations can be made in the quantitative models to simplify the calculational algorithm.
	Cutset merging	The process of cutset development and merging may produce some anomalies depending on the computer code. These include truncation on number of cutsets, and truncation on cutset order.
	Treatment of HFE dependencies	In addition to the determination of HFE dependencies on the model, the manner in which they are treated in the quantification process (e.g. with direct incorporation or with post-processing) could lead to a source of model uncertainty. *The treatment of HFE dependencies have been identified in Table A-2 as a general source of model uncertainty.*
(QU-C) Quantification of dependencies	--	--
(QU-D) Quantification review	--	--
(QU-E) Quantification of uncertainties	Application of the State-of-Knowledge Correlation	The calculation of a true mean value instead of the point estimate calculation from the Boolean logic model.
(LE-A) Plant damage states	PDS definition and grouping	The grouping of accident sequences or cutsets of similar types together for reporting or for transfer to Level 2 evaluations of consequences.
(LE-B) Accident progression contributors	Use of bounding conditions	Impacts associated with events that are assumed to provide bounding characterizations for all of the events within the group. This form of uncertainty includes considerations related to assumptions regarding what is bounding about an event. For LERF evaluations, this includes such things as: 1) potential for assuming guaranteed failures of equipment rather than a realistic survivability assessment, or 2) the lack of credit for auxiliary/reactor building decontamination factor.

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(LE-B) Accident progression contributors (continued)	Applicability of generic severe accident analyses	Plant-specific analysis is generally used in PRAs. If generic analysis is used, its applicability to the plant must be confirmed.
	Dynamic load effects	A number of severe accident phenomena involving failure modes apply severe dynamic loads on containment. These loads have not been rigorously calculated on a plant-specific basis to assess the containment survivability. *One specific aspect of this issue (the treatment of hydrogen combustion in BWR Mark III and PWR ice condenser plants) has been included in Table A-1 as candidate source of model uncertainty.*
	Source term characterization	The source term is characterized by deterministic calculations. These calculations are subjected to a large spectrum of variables including the following: • The computer code used for the calculation and its associated approximations • The treatment of scrubbing and aerosol loss mechanisms • The inclusion of the auxiliary building, reactor building, or outer containment building in the calculation • The number of radionuclides tracked • The number of radionuclide release and states considered
	In-vessel core melt progression	The modeling of in-core melt progression is based primarily on computer simulations that have been benchmarked against a limited set of experiments or events. Examples of some phenomena that can impact the uncertainty include steam explosions and rare containment challenges as well as in-vessel recovery. *The core melt arrest in-vessel issue has been included in Table A-1 as a candidate source of model uncertainty.* *Additionally, the separate in-vessel core melt progression issue of thermally induced failure of hot legs and steam generator tubes has also been included in Table A-1 as a candidate source of model uncertainty.*

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(LE-B) Accident progression contributors (continued)	***Vessel breach***	Vessel breach failure mode and timing are critical phenomena in the evaluation of containment performance, mitigation system performance, and source term. Structural analysis of this failure mode appears to be subjected to wide differences of opinion reflected in various models. Examples of some phenomena that can impact the uncertainty are as follows: • Ex-vessel cooling • Containment flooding • Vessel failure mode • Vessel failure timing *The vessel failure mode and ex-vessel cooling of lower head issues have been included in Table A-1 as candidate sources of model uncertainty.*
	Ex-vessel core melt progression	The modeling of ex-vessel melt progression is based primarily on computer simulations that have been benchmarked against a limited set of experiments or events. Examples of some phenomena that can impact the uncertainty are as follows: • Containment failure mode versus challenge • Containment failure location versus challenge • HPME effects • Impact of water on steel containment/liner failure *The core debris contact with containment issue has been included in Table A-1 as a candidate source of model uncertainty.*

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(LE-B) Accident progression contributors (continued)	Fission product transport, scrubbing, retention, and so on	The source term is characterized by deterministic calculations. These calculations are subjected to a large spectrum of variables including the following: • The computer code used for the calculation and its associated approximations • The treatment of scrubbing and aerosol loss mechanisms (deposition and settling) • The inclusion of the auxiliary building, reactor building, or outer containment building in the calculation • The number of radionuclides tracked • The number of radionuclide release and states considered • The chemistry of released fission products
	Evacuation times	The evacuation times are strong functions of various time-dependent events such as: • Season • Time of day • Weather • Seismic response
	Time in life (core inventory)	Burnup and axial power shape may influence decay heat, time to core damage, fission product inventory.
(LE-C) Accident progression sequences	Severe accident thermal hydraulic codes	The use of an approximate thermal-hydraulic model at a representative time during the fuel cycle is used in the evaluation of system success criteria. The single model is used to represent all plant life. Limitations in the model development include: • Generic versus plant-specific • Time in fuel cycle (axial profile and burn-up) • Sophistication of the model • Ability to model different events (for example, large LOCA, ATWS, loss of FW)
	Performance during severe accident conditions	Identification of the adverse environment is not always effectively included in the modeling of HEPs.

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(LE-C) Accident progression sequences (continued)	***Familiarity with severe accident mitigation guidelines (SAMGs)***	Severe accident mitigation guidelines (SAMGs) provide methods for coping with severe accidents. SAMGs may not be a priority at the plant. Alternatively, they may be implemented only by the TSC, in which case the TSC must be operational to effectively use the SAMGs. The SAMG effectiveness within the PRA model must reflect the manner in which they are implemented at the plant. *Due to its unique nature, note that credit for the Emergency Response Organization (ERO) is included in Table A-1 as a candidate source of model uncertainty. It is included under the Human Reliability Analysis section rather than in the just the LERF analysis section since it could be applicable to more than just the LERF analysis.*
	Impact of severe accident environments	Crew response under LOOP and SBO conditions are subjected to a wide variety of influences that may not be easily captured in the PRA. These include: • Poor lighting • Poor ventilation • High temperature for personnel • Reduced access • Loss of communication • Significant loss of instrumentation • Loss of computerized systems • Extensive competing actions for restoration of power
	ISLOCA CCF	LERF can be controlled by containment bypass sequences. In turn, ISLOCA or BOC events can result in releases if there are CCFs of the isolation valves. This characterization is generally without sufficient data to support the common-cause failure probabilities under these adverse conditions. *The ISLOCA CCF issue is included in Table A-1 as a candidate source of model uncertainty under the ISLOCA IE frequency determination category.*

Table A-4 (continued)
Disposition of Items Originally Appearing in Appendix H of the Technical Basis Document

Generalized High Level Requirement	Topic	Discussion of Issue
(LE-D) Accident progression analyses of containment capability	Reliance on generic containment ultimate pressure capability analyses	The containment structural analysis performed generically is sometimes used to represent a plant-specific situation. This may apply to all of the required analyses or only to a portion of the analysis, for example, dynamic loading.
	Structural analyses	The containment structural analysis performed generically is sometimes used to represent a plant-specific situation. This may apply to all of the required analyses or only to a portion of the analysis, for example, dynamic loading. Plant-specific analysis adds significant confidence to the assessment but still is limited in the failure modes considered. The containment ultimate pressure capability is generally a quasi-static analysis that reflects some of the plant-specific features of the containment. Its use in the PRA may not adequately model the potential weaknesses in the application of the quasi-static analysis to the severe accident core melt progression. Select issues that may not be treated are: • BWRs: shell-induced failure by debris • Containment failure modes (leak versus rupture) • All alignments and containment configurations: Flooded Deinerted Loss of pool • Failure location variation • Failure size variation • Time at temperature • Dynamic loading
(LE-E) LERF quantification	--	--
(LE-F) LERF quantification review	--	--

B

SAMPLE IMPLEMENTATION OF PROCESS FOR CHARACTERIZING THE SOURCES OF MODEL UNCERTAINTY

B.1 Overview

The intent of this appendix is to show a complete model uncertainty issue characterization assessment for a representative BWR Mark II plant. The incorporation of this information into the PRA model documentation is intended to be sufficient to meet the ASME/ANS PRA Standard Supporting Requirements defined in previous sections of this report for QU-E1, QU-E2, QU-E4, QU-F4, and LE-F3. That is, for each applicable model uncertainty item that was shown in Appendix A (i.e., in Table A-1), a plant-specific issue characterization and assessment is provided to fully satisfy the related supporting requirements. Table B-1 illustrates the implementation of this process where the specific supporting requirements that are being treated are clearly identified. This includes the supporting requirements listed above as well as those supporting requirements for documenting the sources of model uncertainty and related assumptions associated with each element (IE-D3, AS-C3, SC-C3, SY-C3, HR-I3, DA-E3, LE-G4, and IF*-B3).

In addition to the assessment for the generic list of candidate model uncertainties, an assessment of plant-specific features and modeling approaches is performed to determine if additional sources of model uncertainty and related assumptions should be incorporated into the list. This assessment is summarized in Section B.2 with the results of the plant-specific identified items incorporated into Table B-2 with the same structure as the generic list of items shown in Table B-1.

The discussion of the standard sensitivity cases recommended in Section 3 and Appendix A of this report for HEPs and CCF values is provided in Section B.3. Recall that these issues were identified as generic high level sources of modeling uncertainty rather than trying to identify all potential sources of model uncertainty associated with these issues since they are generally understood and accepted as areas of uncertainty that can be significant contributors to CDF and LERF.

Finally, Section B.4 summarizes the findings from this sample implementation of the process for characterizing the sources of model uncertainty.

Table B-1
Representative Issue Characterization for Sources of Model Uncertainty (QU-F4 and LE-F3)

Topic (to meet QU-E1)	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made (to meet QU-E2)	Impact on Model (to meet QU-E4)	Characterization Assessment
Initiating Event Analysis (to support meeting IE-D3)						
1. Grid stability	Recently the stability of at least some local areas of the electric power grid has been questioned. The potential duration and complexities of recovery from such events are hard to dismiss. Three different aspects relate to this issue: 1a. LOOP Initiating Event Frequency 1b. Conditional LOOP Frequency 1c. Availability of dc power to perform restoration actions	LOOP sequences including consequential LOOP sequences	NUREG/CR-6890 [11] is used to develop the prior distribution for the LOOP initiator frequency and incorporates four causal categories (Plant centered, Switchyard centered, Grid related, and Weather related). The priors utilize industry data for the plant centered, switchyard centered, and weather LOOP categories; however, region specific grid related LOOP data that is utilized for the prior. A Bayesian update for each category with plant specific data from 2005-2007 is utilized to obtain a total plant specific LOOP frequency.	1. The generic industry data for the four LOOP categories is applicable to the site and appropriate to use as a prior distribution for the plant-specific LOOP frequency development and three years worth of additional plant-specific experience is sufficient to perform the Bayesian update process.	1. The LOOP initiator frequency is apportioned into the four causal factors in the model with a percentage assigned to each category.	The overall approach for the LOOP frequency and fail to recover probabilities utilized is considered an industry good practice, but is not yet considered a consensus model approach.

Table B-1 (continued)
Representative Issue Characterization for Sources of Model Uncertainty (QU-F4 and LE-F3)

Topic (to meet QU-E1)	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made (to meet QU-E2)	Impact on Model (to meet QU-E4)	Characterization Assessment
Initiating Event Analysis (to support meeting IE-D3)						
			The industry wide data in NUREG/CR-6890 [11] for the failure to recovery probabilities for the four LOOP categories are utilized directly for the applicable time frames in the model.	2. The industry-wide recovery data is applicable to the site for the four causal factors included in the model.	2. LOOP recovery failures are included for 0.5, 2.5, 5.0, 10.0, and 20.0 hours from sequence initiation depending on the accident sequence progression.	
			The consequential LOOP failure probabilities are derived consistent with the NRC recommended generic values [12] of ~2E-3 and ~2E-2 given a reactor trip or LOCA, respectively.	3. The use of generic data for consequential LOOP events is assumed to be applicable for the site and the consequential LOOP events are assumed to be similar to other loss of grid events.	3. The loss of grid LOOP recovery failure data is utilized for the consequential LOOP event sequences.	Realistic with slight conservative bias slant on the consequential LOOP probabilities utilized. As such, this should not be a source of model uncertainty in most applications.
			Offsite power restoration is dictated by procedure. Restoration is possible via breaker control using dc power available via separate batteries in the switchyard.	4. When offsite power is available at the switchyard, then power is available to charge the batteries needed for breaker control to align power to the site. The specific failure modes of the offsite restoration are implicitly included via the use of the generic LOOP recovery probabilities.	4. No additional adjustments or system model changes are incorporated when using the different LOOP recovery probabilities. Available recovery times available conservatively chosen to account for restoration time uncertainty.	Realistic with slight conservative bias slant on the recovery times utilized. This should not be a source of model uncertainty in most applications.

Table B-1 (continued)
Representative Issue Characterization for Sources of Model Uncertainty (QU-F4 and LE-F3)

Topic (to meet QU-E1)	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made (to meet QU-E2)	Impact on Model (to meet QU-E4)	Characterization Assessment
2. Support System Initiating Events	Increasing use of plant-specific models for support system initiators (e.g. loss of SW, CCW, or IA, and loss of ac or dc buses) have led to inconsistencies in approaches across the industry. A number of challenges exist in modeling of support system initiating events: 2a. Treatment of common cause failures 2b. Potential for recovery	Support system event sequences	Support System Initiating Event fault trees are developed for loss of SW, loss of IA, loss of RECW, and loss of TECW.	1. The loss of support system success criteria are developed consistent with the post-trip configuration requirements (e.g., 1 of 3 SW pumps) and mission time requirements (i.e., 24 hour MTTR assumed consistent with the 24 hour mitigation mission time).	1. For the standby contributors in the support system initiating event, the same basic events are utilized in the SSIE fault tree and in the mitigation fault tree.	Realistic with slight conservative bias slant because MTTR is typically less than 24 hours. This should not be a source of model uncertainty in most applications.
			The CCF for the fail-to-run terms is based on annualized mission times using generic alpha factors, but with plant-specific information for the independent failure rate.	2. The use of the generic alpha factors based on industry-wide experience is applicable for the site.	2. The fail-to-run CCF terms dominate the overall contribution to the SSIE frequency evaluation.	Slight conservative bias treatment since alpha factors are known to be high when utilized in an annualized fashion and compared to plant-specific experience. This should not be a source of model uncertainty in most applications.
			The support system initiating events are generally used as is with no additional credit for recovery. The exception is that late recovery of IA to support containment venting is credited.	3. The lack of credit for recovery from the support system initiating events will not significantly impact the CDF and LERF distribution.	3. With the exception of the loss of IA recovery term, no basic events included in model for recovery from the loss of support system initiators.	The loss of IA recovery for containment venting is identified as a candidate source of model uncertainty.

Table B-1 (continued)
Representative Issue Characterization for Sources of Model Uncertainty (QU-F4 and LE-F3)

Topic (to meet QU-E1)	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made (to meet QU-E2)	Impact on Model (to meet QU-E4)	Characterization Assessment
3. LOCA initiating event frequencies	It is difficult to establish values for events that have never occurred or have rarely occurred with a high level of confidence. The choice of available data sets or use of specific methodologies in the determination of LOCA frequencies could impact base model results and some applications.	LOCA sequences	The pipe break portion of the LOCA initiating event frequencies are based on a pipe segment count and per segment failure probabilities from the EPRI methodology [15]. The component rupture portion of the LOCA initiating event frequencies are based on the component rupture data and methodology utilized in the NRC RMIEP study [21].	1. The use of generic data from the EPRI methodology and RMIEP study is generally applicable to the site.	1. In general, the LOCA frequencies are higher than those reported in more recent studies (e.g., NUREG-6928 [14]). Therefore, a slight conservative bias in the LOCA initiating event frequencies might be present.	The LOCA frequency values represent a slight conservative bias treatment. This should not be a source of model uncertainty in most applications.
Accident Sequence Analysis (to support meeting AS-C3)						
4. Operation of equipment after battery depletion	Station Blackout events are important contributors to baseline CDF at nearly every US NPP. In many cases, battery depletion may be assumed to lead to loss of all system capability. Some PRAs have credited manual operation of systems that normally require dc for successful operation (e.g., turbine-driven systems such as RCIC and AFW).	Credit for continued operation of these systems in sequences with batteries depleted (e.g., long-term SBO sequences)	No credit is taken for continued operation of any systems without dc power that normally require dc power for operation. This includes HPCI, RCIC, and the SRVs.	1. Operation of systems without dc that normally require dc for operation is not readily viable.	1. Systems that normally require dc for operation are not credited for continued operation upon battery depletion in the event sequence modeling.	No credit for equipment operation after battery depletion may represent a slight conservative bias treatment. This should not be a source of model uncertainty in most applications.

Table B-1 (continued)
Representative Issue Characterization for Sources of Model Uncertainty (QU-F4 and LE-F3)

Topic (to meet QU-E1)	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made (to meet QU-E2)	Impact on Model (to meet QU-E4)	Characterization Assessment
5. RCP seal LOCA treatment – PWRs	The assumed timing and magnitude of RCP seal LOCAs given a loss of seal cooling can have a substantial influence on the risk profile.	Accident sequences involving loss of seal cooling	N/A	N/A	N/A	N/A
6. Recirculation pump seal leakage treatment – BWRs w/ Isolation Condensers	Recirculation pump seal leakage can lead to loss of the Isolation Condenser. While recirculation pump seal leakage is generally modeled, there is no consensus approach on the likelihood of such leaks.	Accident sequences with long-term use of isolation condenser	N/A	N/A	N/A	N/A
Success Criteria (to support meeting SC-C3)						
7. Impact of containment venting on core cooling system NPSH	Many BWR core cooling systems utilize the suppression pool as a water source. Venting of containment as a decay heat removal mechanism can substantially reduce NPSH, even lead to flashing of the pool. The treatment of such scenarios varies across BWR PRAs.	Loss of containment heat removal scenarios with containment venting successful	No credit is taken for the use of injection systems with suction from the suppression pool following containment venting.	1. Upon successful initiation of containment venting, it is assumed that NPSH is lost for all systems taking suction from the suppression pool (i.e., HPCI, RCIC, and LP ECCS – CS and LPCI).	1. HPCI, RCIC, LPCI and Core Spray are not credited for success after containment venting.	No credit for these systems after containment venting represents a slight conservative bias treatment. This should not be a source of model uncertainty in most applications.

Table B-1 (continued)
Representative Issue Characterization for Sources of Model Uncertainty (QU-F4 and LE-F3)

Topic (to meet QU-E1)	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made (to meet QU-E2)	Impact on Model (to meet QU-E4)	Characterization Assessment
8. Core cooling success following containment failure or venting through non hard pipe vent paths	Loss of containment heat removal leading to long-term containment over-pressurization and failure can be a significant contributor in some PRAs. Consideration of the containment failure mode might result in additional mechanical failures of credited systems. Containment venting through "soft" ducts or containment failure can result in loss of core cooling due to environmental impacts on equipment in the reactor building, loss of NPSH on ECCS pumps, steam binding of ECCS pumps, or damage to injection piping or valves. There is no definitive reference on the proper treatment of these issues.	Long-term loss of decay heat removal scenarios	*With containment venting unsuccessful:* Limited credit is taken for continued injection immediately before and after containment failure from the CRD system only. CRD is the only viable injection source as the containment pressure rises above the vent pressure because the SRVs will close on high containment pressures and the RPV will re-pressurize above the low pressure injection capabilities.	1. Low pressure injection sources internal to containment (LPCI and Core Spray from the suppression pool) are assumed to be lost before containment failure when the RPV re-pressurizes above their discharge pressure limits and are also assumed to fail after containment failure due to the items listed in the discussion of the issue.	1. LPCI and Core Spray are not credited for success after containment failure.	No credit for these systems after containment failure may represent a slight conservative bias slant. This should not be a source of model uncertainty in most applications.
				2. HPCI and RCIC are assumed to be unavailable prior to containment failure since high pool temperatures would preclude their use and the RPV would be depressurized per procedure prior to the SRVs re-closing and then after the SRVs can re-open as the containment depressurizes following containment failure.	2. HPCI and RCIC are not credited for success after containment failure.	No credit for these systems after containment failure may represent a slight conservative bias slant. This should not be a source of model uncertainty in most applications.

Table B-1 (continued)
Representative Issue Characterization for Sources of Model Uncertainty (QU-F4 and LE-F3)

Topic (to meet QU-E1)	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made (to meet QU-E2)	Impact on Model (to meet QU-E4)	Characterization Assessment
				3. Other alternate low pressure injection systems are not credited for success since they cannot inject during the time frame prior to containment failure when the SRVs close and the RPV re-pressurizes above their shutoff head.	3. Other alternate low pressure injection systems (e.g., Fire Water and RHRSW through RHR) are also not credited for success after containment failure.	No credit for these systems after containment failure represents a slight conservative bias slant. This should not be a source of model uncertainty in most applications.
				4. Following containment failure, injection from CRD could still be maintained, but if a large containment failure occurs low in the reactor building, CRD is also assumed to be lost. This failure probability is based on a detailed structural analysis of the Mark II containment design.	4. CRD is credited for success after containment failure, but an additional basic event is included that represents the likelihood that the containment failure size and location disrupts the capability of CRD to inject.	CRD injection capability after containment failure is identified as a candidate source of model uncertainty.

Table B-1 (continued)
Representative Issue Characterization for Sources of Model Uncertainty (QU-F4 and LE-F3)

Topic (to meet QU-E1)	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made (to meet QU-E2)	Impact on Model (to meet QU-E4)	Characterization Assessment
			With containment venting successful: As with the containment failure cases, low pressure injection is required for use up to the time of reaching the containment vent pressure. This can be satisfied with LPCI or Core Spray from the suppression pool. If containment venting is successful, then a myriad of low pressure external injection systems could also provide an adequate supply of inventory for RPV makeup following containment depressurization.	1. It is assumed that injection sources from the suppression pool will be lost even though they are more likely to survive since a controlled depressurization could occur in these cases compared to the containment failure cases.	1. LPCI and Core Spray are not credited for success after containment venting.	No credit for these systems after containment venting represents a slight conservative bias treatment. This should not be a source of model uncertainty in most applications.
				2. Potentially viable injection systems post-venting include CRD, Condensate, Fire Water or RHRSW through RHR.	2. Logic is included in the post-venting portion of the event sequence modeling for providing RPV injection from these systems. CRD and Condensate also require make-up to the CST or hotwell, respectively.	Slight conservative bias treatment given that no other systems are credited and fire water and RHRSW are also not credited (see #3 below). This should not be a source of model uncertainty in most applications.
				3. However, since no specific direction is included to line-up the alternate injection systems prior to venting containment, the conditions in the reactor building post-venting are assumed to preclude their use.	3. A separate basic event for lining up alternate injection from fire water or RHRSW prior to venting is included in the model that is currently set to guaranteed fail.	No credit for these systems after containment venting represents a slight conservative bias treatment. This should not be a source of model uncertainty in most applications.

Table B-1 (continued)
Representative Issue Characterization for Sources of Model Uncertainty (QU-F4 and LE-F3)

Topic (to meet QU-E1)	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made (to meet QU-E2)	Impact on Model (to meet QU-E4)	Characterization Assessment
9. Room heatup calculations	Loss of HVAC can result in room temperatures exceeding equipment qualification limits. Treatment of HVAC requirements varies across the industry and often varies within a PRA. There are two aspects to this issue. One involves whether the SSCs affected by loss of HVAC are assumed to fail (i.e. there is uncertainty in the fragility of the components). The other involves how the rate of room heatup is calculated and the assumed timing of the failure.	Dependency on HVAC for system modeling and timing of accident progressions and associated success criteria.	A combination of design basis calculations for technical specifications and Appendix R supporting calculations are referenced to determine the HVAC requirements in the model.	1. Appendix R calculations showing that HVAC is not required for 72 hours in the switchgear and battery rooms are sufficiently applicable to the anticipated transients in the PRA model.	1. An HVAC dependency is not included for the switchgear and battery rooms.	This should not be a candidate source of model uncertainty unless reference to the existing calculations is deemed insufficient.
				2. Core Spray and RHR room cooling is required consistent with the technical specification requirements for these systems.	2. An HVAC dependency for Core Spray and RHR is included in the system models with HVAC failures rendering the system unavailable when demanded.	Realistic with slight conservative bias slant given loss of RHR and CS room cooling failures are assumed to occur at sequence initiation. This should not be a source of model uncertainty in most applications.
				3. The technical specification calculations for HPCI and RCIC are only valid out to six hours. Room cooling is assumed to be required for HPCI/RCIC extended operation beyond six hours.	3. HVAC dependencies for HPCI and RCIC are not included for early operation of these systems, but are included for extended operation beyond six hours.	Realistic with slight conservative bias slant given HPCI and RCIC room cooling may not be needed for the time frame that HPCI and RCIC are utilized. This should not be a source of model uncertainty in most applications.

Table B-1 (continued)
Representative Issue Characterization for Sources of Model Uncertainty (QU-F4 and LE-F3)

Topic (to meet QU-E1)	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made (to meet QU-E2)	Impact on Model (to meet QU-E4)	Characterization Assessment
				4. Appendix R calculations showing that the spray pond pump house fans are not needed for 72 hours as long as doors are opened is judged to be applicable.	4. An HEP for failure to open the SPPH doors is included in the success criteria if the SPPH fans fail to perform their function.	Modeling of the failure to open the SPPH doors should not be a candidate source of model uncertainty unless reference to the existing calculations is deemed insufficient.
				5. Per design basis, two DG exhaust fans are required to operate for each DG cell when the outside ambient air is above 75°F, and only one fan is required when the temperature is below 75°F.	5. A basic event is included in the model representing the likelihood that the ambient temperature is above 75°F for some portion of the DG mission time. This is estimated to be 25% of the time based on engineering judgment.	The percentage of time that two DG HVAC fans are required to provide DG cooling is identified as a candidate source of model uncertainty.
10. Battery life calculations	Station Blackout events are important contributors to baseline CDF at nearly every US NPP. Battery life is an important factor in assessing a plant's ability to cope with an SBO. Many plants only have design basis calculations for battery	Determination of battery depletion time(s) and the associated accident sequence timing and related success criteria.	Design basis calculations indicate that at least 1-3 hours of battery life is available depending on scenario specifics. Credit for 2 hours per division is utilized in the model for scenarios without chargers available.	1. Given the typical conservatisms associated with the design-basis battery calculations, explicit representation of load shedding is not assumed to be required to obtain the 2 or 4 hour battery life times.	1. CRD is only credited in SBO scenarios if both HPCI and RCIC are available to provide initial injection since CRD is not viable as the only makeup source until approximately 4 hours after sequence initiation.	Credit for battery life out to four hours without explicit representation of load shedding is identified as a candidate source of model uncertainty.

Table B-1 (continued)
Representative Issue Characterization for Sources of Model Uncertainty (QU-F4 and LE-F3)

Topic (to meet QU-E1)	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made (to meet QU-E2)	Impact on Model (to meet QU-E4)	Characterization Assessment
	life. Other plants have very plant/condition-specific calculations of battery life. Failing to fully credit battery capability can overstate risks, and mask other potentially contributors and insights. Realistically assessing battery life can be complex.			2. In LOOP events without chargers available, 2 hours is assumed available if only HPCI or RCIC is available; 4 hours of total time is assumed available if both HPCI and RCIC are available.	2. Accounting for RPV inventory boil-off following loss of injection, credit for ac power recovery is included at 0.5, 2.5, or 5.0 hours depending on the availability of HPCI and RCIC.	Realistic with slight conservative bias slant on the times chosen to restore offsite power to avoid core damage is averted following battery depletion. This should not be a source of model uncertainty in most applications.
11. Number of PORVs required for bleed and feed – PWRs	PWR EOPs direct opening of all PORVs to reduce RCS pressure for initiation of bleed and feed cooling. Some plants have performed plant-specific analysis that demonstrate that less than all PORVs may be sufficient, depending on ECCS characteristics and initiation timing.	System logic modeling representing success criterion and accident sequence timing for performance of bleed and feed and sequences involving success or failure of feed and bleed.	N/A	N/A	N/A	N/A

Table B-1 (continued)
Representative Issue Characterization for Sources of Model Uncertainty (QU-F4 and LE-F3)

Topic (to meet QU-E1)	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made (to meet QU-E2)	Impact on Model (to meet QU-E4)	Characterization Assessment
12. Containment sump / strainer performance	All PWRs are improving ECCS sump management practices, including installation of new sump strainers at most plants. There is not a consistent method for the treatment of ECCS sump performance. All BWRs have improved their suppression pool strainers to reduce the potential for plugging. However, there is not a consistent method for the treatment of suppression pool strainer performance.	Recirculation from sump (PWRs) or from the suppression pool (BWRs) system modeling and sequences involving injection from these sources (Note that the modeling should be relatively straightforward, the uncertainty is related to the methods or references used to determine the likelihood of sump strainer and common cause failure of the strainers.)	Individual CCF groups per system are included in the model for the suppression pool suction strainers based on generic and strainer plugging failure data and generic alpha factor data. Additionally, the failure cause and likelihood of suppression pool suction strainers are expected to be significantly different, depending on what type of transient is being analyzed. Therefore, global scenario-specific CCF terms for all suppression pool strainers are also included in the model.	1. A global CCF of all suppression pool strainers (i.e., HPCI, RCIC, 4 CS, and 4 RHR) is highly unlikely, but cannot be totally dismissed. There are different CCF global values utilized for LOCAs (1.0E-5), IORV or emergency depressurization case (1.0E-6), and general transients (1.0E-7) based on engineering judgment. These global failures are assumed to be unrecoverable.	1. Unrecoverable scenario based global CCF terms are utilized in the model for simultaneous failure of all suppression pool strainers.	The incorporation of unrecoverable global CCF term for simultaneous failure of all suppression pool strainers is judged to represent a slightly conservative bias treatment. This should not be a source of model uncertainty in most applications.

Table B-1 (continued)
Representative Issue Characterization for Sources of Model Uncertainty (QU-F4 and LE-F3)

Topic (to meet QU-E1)	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made (to meet QU-E2)	Impact on Model (to meet QU-E4)	Characterization Assessment
13. Impact of failure of pressure relief	Certain scenarios can lead to RCS/RPV pressure transients requiring pressure relief. Usually, there is sufficient capacity to accommodate the pressure transient. However, in some scenarios, failure of adequate pressure relief can be a consideration. Various assumptions can be taken on the impact of inadequate pressure relief.	Success criterion for prevention of RPV overpressure (Note that uncertainty exists in both the determination of the global CCF values that may lead to RPV overpressure and what is done with the subsequent RPV overpressure sequence modeling.)	Failure of a sufficient number of safety relief valves to open when required may lead to excessive reactor vessel pressure and a potential LOCA condition. The success criteria for the reactor pressure control function is established for various scenarios since the number of the relief valves required to open (or relief valve capacity) varies for different accident sequences.	1. For general transients (non-ATWS), it is assumed that 2 of 14 SRVs are required to lift early to preserve RPV integrity below Service Level C. This is conservatively based on the post-trip emergency depressurization success criteria.	1. The actual number of SRVs required to open is insignificant since the dominant failure mechanism is common cause failure of the SRVs where groups of six or more are typically treated as global common cause failures. The global CCF value is based on available generic failure rates and alpha factors.	Slight conservative bias treatment in extension of CCF alpha factors for a group of eight as being applicable to a group of 14. This should not be a source of model uncertainty in most applications.
				2. The success criterion for Large LOCA is suitably equivalent to the impacts of failure of overpressure relief.	2. Transient (non-ATWS) cases with overpressure failures are transferred to the Large LOCA event tree.	Postulated overpressure failure mode being equivalent to Large LOCA success criteria is identified as a candidate source of model uncertainty since the failure mode may be beyond LOCA success criteria capabilities.

Table B-1 (continued)
Representative Issue Characterization for Sources of Model Uncertainty (QU-F4 and LE-F3)

Topic (to meet QU-E1)	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made (to meet QU-E2)	Impact on Model (to meet QU-E4)	Characterization Assessment
				3. Based on plant-specific calculations and reference to generic analysis, it is assumed that 12 of 14 SRVs are required for successful overpressure mitigation in ATWS scenarios.	3. One basic event is included in the model representing the total failure probability that 3 or more SRVs fail to open to provide overpressure protection with the value determined from generic failure rates and alpha factors.	Slight conservative bias treatment in assumption of 100% ATWS conditions for the calculation. This should not be a source of model uncertainty in most applications.
				4. In ATWS scenarios, failure of the vessel pressure relief function is assumed to cause a LOCA that would challenge low pressure ECCS to replenish coolant inventory. The subsequent injection of cold un-borated water under ATWS conditions is assumed to cause re-criticality, eventually leading to core damage.	4. In the ATWS event tree, unmitigated ATWS scenarios with overpressure failure are assigned as core damage sequences.	Slight conservative bias treatment in assumption that overpressure failure in ATWS cases goes directly to core damage. This should not be a source of model uncertainty in most applications.

Table B-1 (continued)
Representative Issue Characterization for Sources of Model Uncertainty (QU-F4 and LE-F3)

Topic (to meet QU-E1)	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made (to meet QU-E2)	Impact on Model (to meet QU-E4)	Characterization Assessment
				5. ARI is assumed to successfully terminate the ATWS event after electrical scram failures, but not before LOCA conditions have occurred if overpressure failures also occur.	5. These sequences are transferred to the Large LOCA event tree for completeness, but are not anticipated to significantly contribute to the Large LOCA frequency given the low probability of occurrence of this exact sequence of events.	Postulated overpressure failure mode being equivalent to Large LOCA success criteria is identified as a candidate source of model uncertainty since the failure mode may be beyond LOCA success criteria capabilities.
Systems Analysis (to support meeting SY-C3)						
14. Operability of equipment in beyond design basis environments	Due to the scope of PRAs, scenarios may arise where equipment is exposed to beyond design basis environments (w/o room cooling, w/o component cooling, w/ deadheading, in the presence of an un-isolated LOCA in the area, etc.).	System and accident sequence modeling of available systems and required support systems	Generally, credit for operation of systems beyond there design-basis environment is not taken. Exceptions are listed in the next column.	1. Appendix R calculations showing that HVAC is not required for 72 hours in the switchgear and battery rooms are sufficiently applicable to the anticipated transients in the PRA model.	1. An HVAC dependency is not included for the switchgear and battery rooms.	This should not be a candidate source of model uncertainty unless reference to the existing calculations is deemed insufficient.
				2. Appendix R calculations showing that the spray pond pump house fans are not needed for 72 hours as long as doors are opened is judged to be applicable.	2. An HEP for failure to open the SPPH doors is included in the success criteria if the SPPH fans fail to perform their function.	Modeling of the failure to open the SPPH doors should not be a candidate source of model uncertainty unless reference to the existing calculations is deemed insufficient.

Table B-1 (continued)
Representative Issue Characterization for Sources of Model Uncertainty (QU-F4 and LE-F3)

Topic (to meet QU-E1)	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made (to meet QU-E2)	Impact on Model (to meet QU-E4)	Characterization Assessment
				3. Per design basis, two DG exhaust fans are required to operate for each DG cell when the outside ambient air is above 75°F, and only one fan is required when the temperature is below 75°F.	3. A basic event is included in the model representing the likelihood that the ambient temperature is above 75°F for some portion of the DG mission time. This is estimated to be 25% of the time based on engineering judgment.	The percentage of time that two DG HVAC fans are required to provide DG cooling is identified as a candidate source of model uncertainty.
				4. Given the typical conservatisms associated with the design-basis battery calculations, explicit representation of load shedding is not assumed to be required to obtain the 2 or 4 hour battery life times.	4. CRD is only credited in SBO scenarios if both HPCI and RCIC are available to provide initial injection since CRD is not viable as the only makeup source until approximately 4 hours after sequence initiation.	Credit for battery life out to four hours without explicit representation of load shedding is identified as a candidate source of model uncertainty.
				5. In LOOP events without chargers available, 2 hours is assumed available if only HPCI or RCIC is available; 4 hours of total time is assumed available if both HPCI and RCIC are available.	5. Accounting for RPV inventory boil-off following loss of injection, credit for ac power recovery is included at 0.5, 2.5, or 5.0 hours depending on the availability of HPCI and RCIC.	Realistic with slight conservative bias slant on the times chosen to restore offsite power to avoid core damage is averted following battery depletion. This should not be a source of model uncertainty in most applications.

Table B-1 (continued)
Representative Issue Characterization for Sources of Model Uncertainty (QU-F4 and LE-F3)

Topic (to meet QU-E1)	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made (to meet QU-E2)	Impact on Model (to meet QU-E4)	Characterization Assessment
Human Reliability Analysis (to support meeting HR-I3)						
15. Credit For ERO	Most PRAs do not give much, if any credit, for initiation of the Emergency Response Organization (ERO), including actions included in plant-specific SAMGs and the new B5b mitigation strategies. The additional resources and capabilities brought to bear via the ERO can be substantial, especially for long-term events.	System or accident sequence modeling with incorporation of HFEs and HEP value determination in both the Level 1 and Level 2 models	Generally, credit for initiation of actions from the ERO is not taken in the Level 1 core damage sequence analysis. Exceptions are noted in the next column. Credit for the SAMGs is taken in the detailed Level 2 analysis. Those impacting the LERF analysis are also listed in the next column.	1. Since containment venting would typically not be directed until 15-20 after sequence initiation given loss of decay heat removal scenarios, a recovery factor on the cognitive portion of the containment vent HEPs include credit for ERO response.	1. Per the EPRI HRA Calculator methodology [22], the cognitive portion of the containment vent HEP is adjusted by 0.1, but the execution portion of the HEP is not adjusted.	Credit for some direction from the ERO for this action is a realistic assumption. Slight conservative bias treatment in the 0.1 factor value utilized. This should not be a source of model uncertainty in most applications.
				2. The use of fire water injection in late time frame scenarios also includes a recovery factor on the cognitive portion of the HEP evaluation.	2. Per the EPRI HRA Calculator methodology [22], the cognitive portion of the fire water injection HEP is adjusted by 0.1, but the execution portion of the HEP is not adjusted. Additionally, however, a separate basic event for lining up alternate injection from fire water or RHRSW prior to venting is included in the model that is currently set to guaranteed fail based on lack of specific procedural direction to do so prior to venting.	Credit for some direction from the ERO for this action is a realistic assumption. Slight conservative bias treatment in the 0.1 factor value utilized. This should not be a source of model uncertainty in most applications.

Table B-1 (continued)
Representative Issue Characterization for Sources of Model Uncertainty (QU-F4 and LE-F3)

Topic (to meet QU-E1)	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made (to meet QU-E2)	Impact on Model (to meet QU-E4)	Characterization Assessment
Internal Flooding (to support meeting IF*-B3)						
16. Piping failure mode	One of the most important, and uncertain, inputs to an internal flooding analysis is the frequency of floods of various magnitudes (e.g., small, large, catastrophic) from various sources (e.g., clean water, untreated water, salt water, etc.). EPRI has developed some data, but the NRC has not formally endorsed its use.	Likelihood and characterization of internal flooding sources and internal flood event sequences	Internal flood analysis and initiating event frequencies for spray, flood, and major flood scenarios developed consistent with the EPRI methodology [B-6].	1. The use of generic flood frequencies with plant-specific estimates of pipe lengths is suitable for representation of the flood frequencies at the site.	1. Flood initiator frequencies are based on plant-specific estimates of pipe lengths and generic flood frequencies (per foot) for different categories of piping from the EPRI methodology [23].	Considered an industry good practice approach, but is not yet a consensus model approach.
				2. Spray flood scenarios with less than 100 GPM flow do not totally disable the system they arise from.	2. Spray initiator scenario impacts are limited to the local affects of the spray.	Realistic with a slight conservative bias slant employed in the undeveloped spray scenarios that are subsumed in with the other flood scenarios in the same region. This should not be a source of model uncertainty in most applications.
				3. Flood and major flood sources are assumed to totally disable the system they arise from.	3. Flood and major flood initiator scenarios include failure of the source system as well as the components that are failed due to the flood event.	Slight conservative bias treatment in that the system may not be totally disable in all cases. This should not be a source of model uncertainty in most applications.

Table B-1 (continued)
Representative Issue Characterization for Sources of Model Uncertainty (QU-F4 and LE-F3)

Topic (to meet QU-E1)	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made (to meet QU-E2)	Impact on Model (to meet QU-E4)	Characterization Assessment
LERF Analysis (to support meeting LE-G4)						
17. Core melt arrest in-vessel	Typically, the treatment of core melt arrest in-vessel has been limited. However, recent NRC work has indicated that there may be more potential than previously credited. An example is credit for CRD in BWRs.	Level 2 containment event tree sequences	In LOOP/SBO events, credit for core melt arrest in-vessel prior to vessel failure is accounted for with adjustments to the LOOP fail to recover values based on representative times between CD and VF.	1. In LOOP/SBO events, credit for core melt arrest in-vessel is based on the following assumed times between core damage and vessel failure: • 0.5 hrs with core damage occurring when offsite power is not recovered in 0.5 hrs. • 1.0 hrs with core damage occurring when offsite power is not recovered in 2.5 hrs. • 1.5 hrs with core damage occurring when offsite power is not recovered in 5.0 hrs. • 2.0 hrs with core damage occurring when offsite power is not recovered in 10.0 hrs.	1. The corresponding differences in the failure to recover probabilities are included in the Level 2 event sequence modeling for LOOP/SBO without offsite power recovered at the time of core damage.	Realistic with slight conservative bias slant on the times chosen to restore offsite power to avoid vessel failure following core damage. This should not be a source of model uncertainty in most applications.

Table B-1 (continued)
Representative Issue Characterization for Sources of Model Uncertainty (QU-F4 and LE-F3)

Topic (to meet QU-E1)	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made (to meet QU-E2)	Impact on Model (to meet QU-E4)	Characterization Assessment
			Only marginal credit for recovery is taken for events that remain at high pressure between core damage and vessel failure.	2. Based on engineering judgment, a factor of 0.9 is assumed to be appropriate for the failure probability to use to credit core melt arrest in-vessel for cases with the RPV remaining at high pressure following core damage.	2. High pressure core damage scenarios with no subsequent RPV depressurization following core damage employ the 0.9 factor for failure to arrest core melt in-vessel in the Level 2 containment event tree sequences.	Core melt arrest in-vessel at high pressure may not be possible and therefore this could be a source of model uncertainty. However, the 0.9 factor compared to the alternative assumption of 1.0 should not have an impact in most applications.
			If RPV depressurization occurs after core damage, then core melt arrest in-vessel is credited if LP ECCS or RHRSW injection is available.	3. Injection from these high capacity low pressure systems will preclude vessel failure if they are available following RPV depressurization given core damage occurs at high RPV pressure.	3. High pressure core damage scenarios with subsequent RPV depressurization following core damage link directly to the LP ECCS and RHRSW injection system fault trees to determine the likelihood of core melt arrest in-vessel.	Core melt arrest prior to vessel failure may not be guaranteed with LP injection recovered after core damage, but prior to vessel failure. Therefore, the assumption of LP ECCS restoration assuring that vessel failure is avoided is identified as a candidate source of model uncertainty.

Table B-1 (continued)
Representative Issue Characterization for Sources of Model Uncertainty (QU-F4 and LE-F3)

Topic (to meet QU-E1)	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made (to meet QU-E2)	Impact on Model (to meet QU-E4)	Characterization Assessment
			If containment failure occurs prior to core damage (LOCA Class 3D and ATWS Class 4) in scenarios that could result in LERF, only injection from RHRSW injection is credited to provide core melt arrest in-vessel.	4. Besides the failure modes of implementing RHRSW injection, additional failure modes are included for harsh reactor building environment (0.25) or piping failures due to containment failure (0.10) based on engineering judgment.	4. Core damage sequences that have LERF potential where containment failure occurs prior to core damage include logic in the core melt arrest in-vessel node for the hardware failures for RHRSW injection and the additional failure modes for harsh reactor building environment or piping failures.	The harsh reactor building environment factor following containment failure of 0.25 and piping failure value of 0.1 following containment failure are both identified as candidate sources of model uncertainty.
18. Thermally induced failure of hot leg/SG tubes – PWRs	NRC analytical models and research findings continue to show that TI-SGTR is more probable than predicted by the industry. There is a need to come to agreement with NRC on the thermal hydraulics modeling of TI SGTR.	Level 2 containment event tree sequences	N/A	N/A	N/A	N/A
19. Vessel failure mode	The progression of core melt to the point of vessel failure remains uncertain. Some codes (MELCOR) predict that even vessels with lower head penetrations will remain intact until the	Level 2 containment event tree sequences	There are four phenomenological conditions that could lead to early containment failure (and LERF) that are dependent upon the vessel failure mode considered in the Level 2 analysis. These four issues are: 1) RPV	1. RPV catastrophic failure leading to early containment failure via missiles or pedestal failure is extremely unlikely based on reference to generic studies and identification of plant-specific features.	1. Failure modes considered in model for missile failures (1E-4) and pedestal failure (1E-6) for sequences that proceed to vessel failure.	Values chosen represent a slight conservative bias treatment given the current understanding of these issues. This should not be a source of model uncertainty in most applications.

Table B-1 (continued)
Representative Issue Characterization for Sources of Model Uncertainty (QU-F4 and LE-F3)

Topic (to meet QU-E1)	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made (to meet QU-E2)	Impact on Model (to meet QU-E4)	Characterization Assessment
	water has evaporated from above the relocated core debris. Other codes (MAAP), predict that lower head penetrations might fail early. The failure mode of the vessel and associate timing can impact LERF binning, and may influence HPME characteristics (especially for some BWRs and PWR ice condenser plants).		catastrophic failure, 2) direct containment heating, 3) ex-vessel steam explosion, and 4) core-melt progression overwhelms vapor suppression capabilities.	2. Direct containment heating only possible for high pressure melt scenarios, but noted as very unlikely in high pressure melt scenarios based on reference to generic studies and identification of plant-specific features.	2. DCH failure mode considered in model (1E-3) for sequences that proceed to vessel failure at high pressure.	Values chosen represent a slight conservative bias treatment given the current understanding of these issues. This should not be a source of model uncertainty in most applications.
				3. Ex-vessel steam explosions noted as very unlikely based on reference to generic studies and identification of plant-specific features.	3. Ex-vessel steam explosion failure mode considered in model (1E-3) for sequences that proceed to vessel failure.	Values chosen represent a slight conservative bias treatment given the current understanding of these issues. This should not be a source of model uncertainty in most applications.

Table B-1 (continued)
Representative Issue Characterization for Sources of Model Uncertainty (QU-F4 and LE-F3)

Topic (to meet QU-E1)	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made (to meet QU-E2)	Impact on Model (to meet QU-E4)	Characterization Assessment
				4. Ex-vessel core melt progression overwhelms vapor suppression noted as extremely unlikely for low pressure RPV failures modes and very unlikely for high pressure failure modes based on reference to generic studies and identification of plant-specific features. However, more recent MAAP results indicate that early failure of the drain pot in the pedestal region (leading to pool bypass shortly after vessel failure) with subsequent flow of core debris into the suppression pool might hinder the vapor suppression capabilities of the site more than what was originally considered.	4. Ex-vessel core melt progression overwhelms vapor suppression considered in model for low pressure RPV failure sequences (1E-5) and high pressure RPV failure sequences (1E-3).	Ex-vessel core melt progression overwhelms vapor suppression capabilities is identified as a potential candidate source of model uncertainty.

Table B-1 (continued)
Representative Issue Characterization for Sources of Model Uncertainty (QU-F4 and LE-F3)

Topic (to meet QU-E1)	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made (to meet QU-E2)	Impact on Model (to meet QU-E4)	Characterization Assessment
20. Ex-vessel cooling of lower head	The lower vessel head of some plants may be submerged in water prior to the relocation of core debris to the lower head. This presents the potential for the core debris to be retained in-vessel by ex-vessel cooling. This is a complex analysis impacted by insulation, vessel design and degree of submergence.	Level 2 containment event tree sequences	Containment flooding is procedurally directed in most core damage scenarios. However, given the Mark II containment design, no credit is taken for flooding containment in time to prevent vessel failure via ex-vessel cooling of the lower RPV head. Incorporation of containment flooding is only included in the full Level 2 model to differentiate some of the non-LERF release categories.	1. Ex-vessel cooling of the lower head cannot occur quickly enough to prevent vessel failure and the potential for LERF scenarios.	1. Ex-vessel cooling of the lower head is not included in the model.	No credit for ex-vessel cooling of the lower head represents a realistic treatment with a slight conservative bias slant. This should not be a source of model uncertainty in most applications.
21. Core debris contact with containment	In some plants, core debris can come in contact with the containment shell (e.g., some BWR Mark Is, some PWRs including free-standing steel containments). Molten core debris can challenge the integrity of the containment boundary. Some analyses have demonstrated that core debris can be cooled by overlying water pools.	Level 2 containment event tree sequences	There are some postulated failure modes that could result in some debris reaching the Mark II liner. However, because the structural member of containment is the concrete, it is judged that small amounts of debris adjacent to the liner will not compromise containment integrity.	This issue is ruled out and not developed further.	N/A	N/A

Table B-1 (continued)
Representative Issue Characterization for Sources of Model Uncertainty (QU-F4 and LE-F3)

Topic (to meet QU-E1)	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made (to meet QU-E2)	Impact on Model (to meet QU-E4)	Characterization Assessment
22. ISLOCA IE Frequency Determination	ISLOCA is often a significant contributor to LERF. One key input to the ISLOCA analysis are the assumptions related to common cause rupture of isolation valves between the RCS/RPV and low pressure piping. There is no consensus approach to the data or treatment of this issue. Additionally, given an overpressure condition in low pressure piping, there is uncertainty surrounding the failure mode of the piping.	ISLOCA initiating event sequences	Detailed ISLOCA analysis includes the relevant considerations listed in IE-C12 of the ASME/ANS PRA Standard and accounts for common cause failures and captures likelihood of different piping failure modes.	1. Common cause beta factors from NUREG/CR-5497 [24] are utilized for the MOVs and CVs that comprise potential ISLOCA pathways.	1. One ISLOCA initiating event frequency is implemented in the model representing the sum of all of the individual flow paths analyzed for rupture initiating event frequency.	The approach for the ISLOCA frequency determination is considered an industry good practice, but is not yet considered a consensus model approach.
				2. The failure probability for each flow path given exposure to high pressure RPV conditions is appropriately represented by the formulae in NUREG/CR-5603 [25].	2. Unique contributions from each flow path included in the model via a multiplier on the total ISLOCA initiating event frequency to delineate that fraction of system unavailability from the initiating event.	
23. Treatment of Hydrogen combustion in BWR Mark III and PWR ice condenser plants	The amount of hydrogen burned, the rate at which it is generated and burned, the pressure reduction mitigation credited by the suppression pool, ice condenser, structures, etc. can have a significant impact on the accident sequence progression development.	Level 2 containment event tree sequences	N/A	N/A	N/A	N/A

B.2 Consideration of Plant-Specific Features/Modeling Approaches

This portion of the assessment allows for the identification of any plant-specific features that require unique phenomenological assessments not considered in the generic list in Appendix A.

For the representative BWR Mark II plant utilized in this sample assessment, a review of all unique phenomenological data assessments was performed to identify the following items as plant-specific candidate sources of model uncertainty:

- Probability that flooding of the steam line fails all SRVs
- Diesel generator failure to repair probabilities
- RHR, RHRSW, and ESW pump repair failure probabilities
- Probability of containment integrity challenge following vessel rupture event
- Digital feedwater control failure probabilities

Each of these items is discussed in Table B-2 in the same format as that utilized in Table B-1 to provide an issue characterization for the plant-specific sources of model uncertainty.

Table B-2
Representative Issue Characterization for Additional Plant-Specific Sources of Model Uncertainty (QU-F4 and LE-F3)

Topic (to meet QU-E1)	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made (to meet QU-E2)	Impact on Model (to meet QU-E4)	Characterization Assessment
Consideration of Plant-Specific Features / Modeling Approaches						
Potential for inadvertent flooding of steam lines to fail the SRVs	The steam lines in BWRs may become flooded if level is not maintained below Level 8, the automatic trip functions fail, and operators do not respond in time to avoid the steam lines from becoming flooded. The water in the steam lines could then disable the SRVs from being able to perform their pressure control function even if the RPV water level drops later.	System logic model for use of SRVs for depressurization	Specific modeling is not included for the feedwater system, but is included for HPCI and RCIC. Given the conditions occur that would allow uncontrolled flooding of the steam lines, a probability is assigned that this uncontrolled flooding permanently disables all of the SRVs.	1. Likelihood of failure of FW automatic control, followed by failure of manual control, followed by failure of all Level 8 trips, followed by failure of all SRVs deemed sufficiently unlikely such that it is not needed to be explicitly model.	1. Feedwater failures leading to flooded SRVs not explicitly modeled.	Likely acceptable to be screened but could be important in some applications of the model.
				2. Based on training and simulator observations, operators are expected to take early manual control of HPCI/RCIC.	2. A human error event for failure to take manual control of HPCI/RCIC early is included in the model.	This event is encompassed within the identification of human error probabilities as a general source of model uncertainty.
				3. If early manual control fails and subsequent Level 8 trips fail, then separate actions must be taken to terminate or control flow prior to flooding the steam lines. The time available would be significantly different depending on whether HPCI or RCIC fails to trip on Level 8.	3. Separate human error events are included in the model for failure to take control prior to flooding the steam lines given RCIC Level 8 trip fails or HPCI Level 8 trip fails.	This event is encompassed within the identification of human error probabilities as a general source of model uncertainty.

Table B-2 (continued)
Representative Issue Characterization for Additional Plant-Specific Sources of Model Uncertainty (QU-F4 and LE

Topic (to meet QU-E1)	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made (to meet QU-E2)	Impact on Model (to meet QU-E4)	Characterization Assessment
Consideration of Plant-Specific Features / Modeling Approaches						
				4. If the steam lines become flooded in an uncontrolled fashion, then there is some likelihood that the SRVs could fail to perform their depressurization function later.	4. Although the SRVs are designed to pass water, they are never tested in this fashion. A nominal 5% failure probability is assigned.	Flooding of the steam lines leading to failure of the SRVs is identified as a candidate source of model uncertainty.
Diesel generator repair probabilities	There are model uncertainties related to the applicability of available data sources for determining the diesel generator repair probabilities	LOOP sequences with diesel generator failures	The diesel generator repair probabilities are included in the model at various time frames consistent with the time frames utilized for LOOP recovery.	1. The diesel generator repair probabilities derived from selected past studies are deemed to be applicable for the site.	1. Diesel generator repair probabilities are included in the final sequence cutset results.	Credit for diesel generator repair is identified as a candidate source of model uncertainty.
RHR, RHRSW, ESW pump repair failure probabilities	There are model uncertainties related to the applicability of available data sources for determining the pump repair probabilities	System logic models for RHR, RHRSW, and ESW	The pump repair probabilities are included in the model for long time frame sequences. Different probabilities are utilized for LOCAs versus transients because of the different times available.	1. The pump repair probabilities derived from selected past studies are deemed to be applicable for the site.	1. RHR, RHRSW, and ESW pump repair probabilities are included in the final sequence cutset results.	Credit for RHR, RHRSW, and ESW pump repair is identified as a candidate source of model uncertainty.

Table B-2 (continued)
Representative Issue Characterization for Additional Plant-Specific Sources of Model Uncertainty (QU-F4 and LE

Topic (to meet QU-E1)	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made (to meet QU-E2)	Impact on Model (to meet QU-E4)	Characterization Assessment
Containment integrity following vessel rupture event	There is model uncertainty regarding the subsequent treatment that increases the likelihood of LERF for this extremely rare event.	Vessel rupture sequences	The Level 2 sequence modeling differentiates those vessel rupture scenarios that also result in early containment failure.	1. A portion of the vessel rupture sequences are assumed to result in concurrent containment failure coincident with the vessel rupture.	1. The scenarios that result in early containment failure are classified as accident class 3D scenarios with a high potential for LERF.	Containment integrity following vessel rupture is identified as a candidate source of model uncertainty.
Digital feedwater control failure probabilities	There are model uncertainties associated with modeling digital systems, such as those related to determining the failure modes of these systems and components.	Feedwater system logic model	The reliability values from the vendor study demonstrating that the system performance would result in less than 0.1 transients per year are used for the key components of the system.	1. The reliability analysis for causing plant trips performed by the vendor is assumed to be equally applicable to the reliability of the system post plant trips that are caused by other means that do not directly affect the feedwater availability.	1. Basic events representing the reliability values for the auto level controller, the field buses, false signal from the redundant reactivity control system, and false signal from the Level 8 trip system are included in the system logic model.	Digital feedwater control failure probabilities are identified as a candidate source of model uncertainty.

B.3 HEP and CCF Sensitivity Case Results

The process outlined in Section 3 and Appendix A of this report also included the recommendation to perform a standard set of four sensitivity cases:

- All HEP probabilities (including pre-initiators, post-initiators, and dependent HEP values) set to their 5th percentile value (the use of zero-value HEP probabilities is also deemed acceptable)
- All HEP probabilities (including pre-initiators, post-initiators, and dependent HEP values) set to their 95th percentile value
- All CCF probabilities set to their 5th percentile value (the use of zero-value CCF probabilities is also deemed acceptable)
- All CCF probabilities set the their 95th percentile value

The results of these sensitivity cases are also discussed here and compared to the RG 1.174 CDF and LERF limits of $1x10^{-4}$/yr for CDF and $1x10^{-5}$/yr for LERF to obtain insights into the sensitivity of the base PRA model results to these generic high level sources of modeling uncertainty rather than trying to identify all potential sources of model uncertainty associated with these issues since they are generally understood and accepted as areas of uncertainty that can be significant contributors to CDF and LERF. The results of these sensitivity studies are presented in Table B-3. Only adjustments to the basic events in the base cutset equation were made to determine these results. This is judged to be acceptable for the initial base sensitivity cases.

Table B-3
Formulation of Sensitivity Studies for the Representative Mark II Plant

Sensitivity Study Item	Base Value	Lower Bound or 5th Percentile Value	Upper Bound or 95th Percentile Value	CDF/LERF when at Lower Bound Value	CDF/LERF when at Upper Bound Value
Generic Sensitivity Study Issues (Base CDF = 3.70E-6, Base LERF = 6.80E-8)					
All Human Error Probability Values	Various	Various (1)	Various (1)	1.06E-6 (0.29x Base CDF) 5.11E-8 (0.75x Base LERF)	2.10E-5 (5.7x Base CDF) 1.78E-7 (2.6x Base LERF)
All Common Cause Failure Probability Values	Various	Various	Various	2.67E-6 (0.72x Base CDF) 5.89E-8 (0.87x Base LERF)	6.25E-6 (1.7x Base CDF) 8.99E-8 (1.3x Base LERF)

The results of the special sensitivity studies lead to the following conclusions for this representative Mark II plant:

Human Error Probability Values

- More than 70% of the CDF and about 25% of the LERF base case values include human error terms as contributors.
- Correspondingly, setting all of the HEP values to their 95th percentile values increases the CDF by a factor of 5.7 and LERF by a factor of 2.6.
- However, both CDF and LERF are below the RG-1.174 CDF and LERF limits of $1x10^{-4}$/yr for CDF and $1x10^{-5}$/yr for LERF when all of the HEP values are set to their 95th percentile values.

Common Cause Failure Probability Values

- More than 25% of the CDF and more than 10% of the LERF base case values include human error terms as contributors.
- Correspondingly, setting all of the CCF values to their 95th percentile values increases the CDF by a factor of 1.7 and LERF by a factor of 1.3.
- However, both CDF and LERF are still well below the RG-1.174 CDF and LERF limits of $1x10^{-4}$/yr for CDF and $1x10^{-5}$/yr for LERF when all of the CCF values are set to their 95th percentile values.

The results of these sensitivity cases indicate that human errors (or the potential for human errors) will be potential candidate sources of model uncertainty for many applications of the model. The specific HEPs that contribute will have to be examined on a case by case basis for the application. Similarly, common cause failures (or the potential for common cause failures) will also be potential candidate sources of model uncertainty for many applications of the model and the specific CCF values that contribute will have to be examined on a case by case basis. Given the relative magnitude of the contributions from the sensitivity cases presented here, it is likely that specific HEP values will be more important than specific CCF values for any given application.

B.4 Summary

The results of implementing the process as shown in Tables B-1 and B-2 and discussed in Section B.3 identified the following issues as the most likely candidate sources of model uncertainty from the base PRA model assessment.

- LOOP frequency and fail to recover probabilities
- The loss of IA recovery for containment venting
- CRD injection capability after containment failure
- The percentage of time that two DG HVAC fans are required to provide DG cooling

- Credit for battery life out to four hours without explicit representation of load shedding
- Postulated overpressure failure mode being equivalent to large LOCA success criteria
- Internal flood initiating event frequencies and failure modes
- LP ECCS restoration after core damage assuring that vessel failure is avoided
- Factors that harsh reactor building environment or piping failures fail all injection following containment failure
- Ex-vessel core melt progression overwhelms vapor suppression capabilities
- ISLOCA frequency
- Potential for inadvertent flooding of steam lines to fail the SRVs
- Diesel generator repair probabilities
- RHR, RHRSW, ESW pump repair failure probabilities
- Containment integrity following vessel rupture event
- Digital feedwater control failure probabilities
- Human error probability values
- Dependent human error probability values
- Common cause failure values

For the most part, the issues listed above plus the topics identified in Table A-3 would need to be considered when trying to identify potential sources of model uncertainty relevant to the application being investigated per the guidance provided in Section 4 (see Figure 4-1) in this report. Several other issues were also identified as being treated with conservative bias slants. These items are indicated in Table B-1, but due to the conservative treatment, they should not become potential candidates as key sources of uncertainty for most applications of the model.

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