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Joint Nuclear Regulatory Commission and Electric Power Research Institute Workshop on the Treatment of Probabilistic Risk Assessment Uncertainties

Manuscript Completed: October 2012

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ABSTRACT

A co-sponsored workshop on the treatment of probabilistic risk assessment (PRA) uncertainties was held in Rockville, MD, on February 29 – March 1, 2012, by the United States Nuclear Regulatory Commission (NRC) and the Electric Power Research Institute (EPRI). The purpose of the workshop was to bring together experts to gain a better understanding of the sources of uncertainty, how they are manifested in the PRA, and their potential significance to the PRA model and results. More specifically, the workshop addressed uncertainties associated with risk assessments for internal fires, seismic events, low power and shutdown (LPSD) conditions, and for the Level 2 portion of PRAs. Invited subject matter experts in each of the four topic areas were asked to give a presentation on the first day. These presentations served as a catalyst for group discussion amongst the workshop participants on the first and second days of the workshop. As the individual sessions discussed sources of PRA uncertainty, each topic discussed was categorized as model uncertainty, completeness uncertainty, level of detail uncertainty, or parameter uncertainty, and a subjective significance ranking was assigned to each of HIGH, MEDIUM or LOW. The total number of individual uncertainty issues raised in each topical session was as follows: 59 issues for internal fire, 22 issues for seismic events, 22 issues for LPSD, and 30 issues for Level 2. It appears for seismic events and Level 2 that model uncertainty was the predominant source of uncertainty, while internal fire and LPSD had a more even spread among the various sources of uncertainty. Of the 133 total issues identified among all the topical sessions, 78 issues are expanded in greater detail in the body of this report. These topics were determined to be of MEDIUM or greater significance (occasionally a topic ranked LOW is discussed based on the discretion of the session facilitator). For each of these topics, the following information is presented: 1) a description of the issues or sources of uncertainty, 2) how the issues are manifested in the PRA, 3) a discussion of how the issues are relevant to the base PRA, application, or both, if the issues are applicable to new, existing, or advanced reactors, and the significance ranking (HIGH, MEDIUM, or LOW) for that issue as related to the Standard or draft Standard technical element, and 4) a discussion of potential research and development (R&D) work which may be needed to resolve the issues or uncertainties.

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ACKNOWLEDGEMENTS

The authors would like to recognize the contributions of the individual workshop session moderators and note takers who summarized the findings presented in this report. The principal technical authors of this report, in alphabetical order, were:

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Jeff LaChance (Sandia National Laboratories)

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Gareth Parry (ERIN Engineering)

Don Vanover (ERIN Engineering)

Tim Wheeler (Sandia National Laboratories)

Additionally, the workshop would not have been possible without the subject matter experts, invited on behalf of the Electric Power Research Institute and the Nuclear Regulatory Commission, whose technical presentations were crucial to deriving the sources of model uncertainty in Fire, Seismic, Low Power and Shutdown, and Level 2 PRA.

ACRONYMS AND ABBREVIATIONS

AOP	Abnormal Operating Procedures
AOT	Allowed Outage Time
BNL	Brookhaven National Laboratory
BWR	Boiling Water Reactor
CDF	Core Damage Frequency
CDFM	Conservative Deterministic Failure Margin
CET	Containment Event Tree
CFD	Computational Fluid Dynamics
DOE	Department of Energy
EOF	Emergency Operations Facility
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
EDMG	Extreme Damage Mitigation Guideline
FPRA	Fire Probabilistic Risk Assessment
FSS	Fire Scenario Selection
GMC	Ground Motion Characterization
GMPE	Ground Motion Predication Equation
HCLPF	High Confidence Low Probability of Failure
HEP	Human Error Probability
HFE	Human Failure Event
HRA	Human Reliability Analysis
HRR	Heat Release Rate
HVAC	Heating, Ventilation, and Air Conditioning
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
LPSD	Low Power and Shutdown
MCB	Main Control Board
MSIV	Main Steam Isolation Valve
NPSH	Net Positive Suction Head
NRC	U.S. Nuclear Regulatory Commission
PORV	Pilot Operated Relief Valve
POS	Plant Operational (Operating) States
PRA	Probabilistic Risk Assessment
PSF	Performance Shaping Factor
PSHA	Probabilistic Seismic Hazard Analysis
PWR	Pressurized Water Reactor
QLRA	Qualitative Risk Assessment
R&D	Research and Development
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
RV	Reactor Vessel
SAMG	Severe Accident Management Guidelines

SD	Shutdown
SDP	Significance Determination Process
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SNL	Sandia National Laboratories
SOKC	State of Knowledge Correlation
SRV	Safety Relief Valve
SSC	Seismic Source Characterization
SSCs	Structures, Systems and Components
TSC	Technical Support Center

1. INTRODUCTION

This document presents the outcomes of the United States Nuclear Regulatory Commission (NRC) and Electric Power Research Institute (EPRI) co-sponsored Workshop on the Treatment of PRA Uncertainties. The workshop was held at the Legacy Hotel and Conference Center in Rockville, MD, on February 29 – March 1, 2012. The workshop's purpose was to bring together experts to gain a better understanding of the sources of uncertainty, how they are manifested in Probabilistic Risk Assessments (PRAs), and their potential significance to the PRA model and results. More specifically, the workshop addressed uncertainties associated with risk assessments for internal fires, seismic events, low power and shutdown (LPSD) conditions, and for the Level 2 portion of PRAs.

On the morning of the first day, the workshop and its participants were divided among four parallel sessions (Internal Fires, Seismic Events, LPSD, and Level 2), where invited subject matter experts provided presentations on their perspectives regarding the sources of model uncertainty for their respective areas. A list of invited subject matter expert presenters, the session in which they presented, and their affiliation can be found below in Table 1.

Table 1. Invited Subject Matter Experts

Name	Session	Affiliation
Ray Gallucci	Internal Fire	U.S. Nuclear Regulatory Commission
Brian Metzger	Internal Fire	U.S. Nuclear Regulatory Commission
Paul Guymer	Internal Fire	Jacobson Analytics
Mike Wright	Internal Fire	Jacobson Analytics
Mardy Kazarians	Internal Fire	Kazarians and Associates
Dennis Henneke	Internal Fire	General Electric-Hitachi
Annie Kammerer	Seismic Events	U.S. Nuclear Regulatory Commission
Jim Xu	Seismic Events	U.S. Nuclear Regulatory Commission
M.K. Ravindra	Seismic Events	MKRavindra Consulting
Greg Hardy	Seismic Events	Simpson Gumpertz & Heger
Ken Kiper	LPSD	NextEra Energy
Don Wakefield	LPSD	ABS Consulting
Marie Pohida	LPSD	U.S. Nuclear Regulatory Commission
Steve Eide	LPSD	Scientech
Don Helton	Level 2	U.S. Nuclear Regulatory Commission
Richard Denning	Level 2	The Ohio State University
Mark Leonard	Level 2	dycoda LLC
Jeff Gabor	Level 2	ERIN Engineering
Ray Schneider	Level 2	Westinghouse

Session moderators, representing both NRC and EPRI, were assigned to each session to facilitate the group discussion and collect session findings. Additionally, note-takers were assigned to each session in order to chronicle the findings and conclusions of the group. The session moderators, note-takers, and their affiliation are as follows:

- **Internal Fire**: Jeff LaChance (moderator, Sandia National Laboratories (SNL)), Nick Melly (note-taker, NRC)
- **Seismic Events**: John Lehner (facilitator, Brookhaven National Laboratory (BNL)), Michelle Gonzalez (note-taker, NRC), Brian Wagner (note-taker, NRC)
- **LPSD**: Gareth Parry (facilitator, ERIN Engineering), Matt Dennis (facilitator, SNL), Alysia Bone (note-taker, NRC)
- **Level 2**: Don Vanover (facilitator, ERIN Engineering), Tim Wheeler (facilitator, SNL), Sandra Lai (note-taker, NRC)

In the afternoon, the four sessions remained in parallel and an open discussion among the invited experts and other participants commenced. Each session discussed the details of the individually identified sources of model uncertainty to understand how the sources manifest themselves in the PRA and their impact on the results, what are the issues associated with the sources, and what is the significance of each source.

On the morning of the second day, the workshop participants and expert presenters gathered to hear and discuss summary presentations on findings and conclusions from each session (Internal Fire, Seismic Events, LPSD, and Level 2).

2. REPORT FORMAT

The workshop findings and conclusions are divided into four sections, one for each topical session. For each uncertainty identified during the workshop sessions, it was necessary to

1) Categorize the significance of the identified uncertainty.
2) Determine which uncertainty category (model, completeness, level of detail, or parameter) best described the identified uncertainty issue.
3) If applicable, determine what technical element the uncertainty most closely related to in the Standard or draft Standard for that topic area.

In order to facilitate the categorization of uncertainty significance, the expert presenters were given the following significance determination scheme, which was also used to reclassify and ultimately determine the significance of all uncertainties identified in this report:

1) HIGH = The uncertainty has a moderate to high impact on the conclusions and risk insights.
2) MEDIUM = The uncertainty has a small to moderate impact on the conclusions and risk insights.
3) LOW = The uncertainty has a negligible to small impact on the conclusions and risk insights.

Within each session section, a table is provided listing the technical elements applicable to that subject area versus the categories of uncertainty (model, completeness, level of detail and parameter). Included in the table is the number of uncertainties identified for each significance level (HIGH, MEDIUM, and LOW). It should be noted that some sessions did not identify all categories of uncertainty; therefore, discussion on that category may be omitted from the session section.

2.1 Uncertainty Categories

A majority of the sessions had expert presentations that focused not only on model uncertainty, but also included completeness, level of detail and parameter uncertainty. Because the primary objective of the workshop was to focus on sources of model uncertainty, it is useful to define each type of uncertainty so that the uncertainties raised in each session can be appropriately categorized.

2.1.1 PRA Model Uncertainties

As discussed in NUREG-1855 (NRC, March 2009), model uncertainty is related to an issue for which no consensus approach or model exists 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, and introduction of a new initiating event). A model uncertainty results from a lack of knowledge of how structures, systems and components (SSCs) behave under the conditions arising during the development of an accident. A model uncertainty can arise for the following reasons:

- The phenomenon being modeled is itself not completely understood (e.g., behavior of gravity-driven passive systems in new reactors, or crack growth resulting from previously unknown mechanisms).
- For some phenomena, some data or other information may exist, but they need to be interpreted to infer behavior under conditions different from those in which the data were collected (e.g., Reactor Coolant Pump (RCP) seal Loss of Coolant Accident (LOCA) information).
- The nature of the failure modes is not completely understood or is unknown (e.g., digital instrumentation and controls).

2.1.2 PRA Completeness Uncertainty

Completeness uncertainty is caused by the limitations in the scope of the model, such as whether all applicable physical phenomena have been adequately represented, and/or all accident scenarios that could significantly affect the determination of risk have been identified.

Completeness uncertainty also can be thought of as a type of model uncertainty. However, completeness uncertainty is discussed separately because it represents a type of uncertainty that cannot be quantified and because it represents those aspects of the system that are, either knowingly or unknowingly, not addressed in the model (NRC, March 2009).

2.1.3 PRA Level of Detail Issues

The level of detail generally refers to the level to which a system is modeled (e.g., function level, train level, component level), the extent to which systems are included in the success criteria (e.g., safety systems and non-safety systems), the degree to which events or sequences are subsumed, the extent to which phenomena are included in the challenges to the plant in the

Level 2 analysis, and the extent to which operator actions are considered (e.g., accident management strategies).

Level of detail is generally dictated by four factors: (1) the level of detail to which information is available, (2) the level of detail so that dependencies are included, (3) the level of detail so that the risk contributors are included, and (4) the level of detail sufficient to support the application.

2.1.4 PRA Parameter Uncertainties

Parameter uncertainty is the uncertainty in the values of the parameters of a model and is typically represented by a probabilistic distribution. Examples of parameters that could be uncertain include initiating event frequencies, component failure rates and probabilities, and human error probabilities that are used in the quantification of the accident sequence frequencies.

2.2 Session Findings and Conclusions Format

In this report, the various identified sources of uncertainty are grouped in the proceeding sections according to which type of uncertainty they were identified as mostly closely representing (model, completeness, level of detail, or parameter). As discussed previously, the technical elements of each topical area (Internal Fire, Seismic Events, LPSD, and Level 2) are presented in a table along with the uncertainty categories and number of uncertainties identified for each significance level. As a contrast, Appendices A through D for Internal Fire, Seismic Events, LPSD, and Level 2, respectively, present the sources of uncertainty grouped by the technical element for the respective Standard or draft Standard.

For each session, the HIGH, MEDIUM, and LOW significance sources of model uncertainty are discussed in this report. The sections on completeness, level of detail, and parameter uncertainty capture the portion of issues which were screened as NOT model uncertainties and are thus less rigorously discussed here because the workshop was supposed to capture model uncertainties. Therefore, only those completeness, level of detail, and parameter uncertainties identified as HIGH are discussed here, except in the cases of Seismic Events and LPSD where no HIGH classifications were identified. An exhaustive list of all sources of uncertainty and their significance can be found in Appendices A through D.

For each source of uncertainty, the following information is summarized from discussions during the session or from the general workshop session on the second day:

1. Description of Issue: a description of the issues or sources of uncertainty are given.
2. Manifestation in PRA: a discussion of how the issues are manifested in the PRA.
3. Relevance: a discussion of how the issues are relevant to the base PRA, application or both, if the issues are applicable to new, existing, or advanced reactors, and the significance ranking (HIGH, MEDIUM, or LOW) for that issue as related to the Standard or draft Standard technical element.
4. Potential R&D: a discussion of potential research and development (R&D) work which may be needed to resolve the issues or uncertainties.

Recommendations for potential R&D is optional based on discussion during the session; therefore, it is not always included for each uncertainty identified.

3. UNCERTAINTIES IDENTIFIED FOR INTERNAL FIRE PRA

3.1 Introduction

During the Fire PRA session of the NRC/EPRI Workshop on PRA Uncertainties, six experts from both industry and the NRC presented on sources of PRA uncertainty. The expert presenters followed various formats, which complicated assimilating the viewpoints on different aspects of the Fire PRA (FPRA) model uncertainties. One presenter followed the suggested format using the PRA Standard technical elements as a guide, others reflected on sources of uncertainty with respect to the core damage frequency (CDF) risk equation, while another provided examples of parameter (e.g., heat release rate (HRR), cable damage temperature, etc.) uncertainty evaluation on fire growth. Ultimately, the session presenters and participants decided to categorize the sources of uncertainty based on the PRA Standard. Unlike other sessions at the workshop, the majority of FPRA uncertainty issues raised were related to model uncertainties, which was the ultimate focus of the workshop. Only one presentation, in fact, identified level of detail issues in addition to modeling issues.

Using the high level requirements and technical elements of the Level 1 Internal Fire PRA Standard (ASME/ANS, 2009) as a guide, the FPRA session discussed the sources of model uncertainty for each technical element and ranked those sources on as HIGH, MEDIUM, or LOW. However, several experts indicated that an alternative ranking method may have been more applicable, but one was not established. Within the established ranking and technical element framework, the session identified that the majority of sources of model uncertainty were in the Fire Scenario Selection (FSS) Technical Element of the PRA Standard.

In addition to sources of model uncertainty and importance ranking, each session was asked to identify any unique aspects of the model uncertainty as applied to new or advanced reactors, and any areas which could benefit from additional research and development (R&D) by industry or the NRC. Ultimately, the session did not identify any unique advanced reactor issues, and several issues which could benefit from additional R&D were identified and are discussed in proceeding sections with respect to that topic.

Because the Fire PRA has a longer history and more established criteria in regularly updated consensus standards than other topics such as Level 2 or Low Power and Shutdown (LPSD) PRA, a majority of the discussion in the session centered on sources of model uncertainty.

The sources of uncertainty identified are summarized in Table A-1 in Appendix A categorized by the technical elements of a Fire PRA as defined in the Level 1 Internal Fire PRA Standard (ASME/ANS, 2009). Table 2 in Section 3.2 presents the number of issues identified during the FPRA session. The table presents a list of Fire PRA technical elements versus the type of issue raised (model uncertainty, completeness, level of detail, or parameter uncertainty) and shows the number of issues raised in a HIGH, MEDIUM or LOW significance category which the session moderator, expert presenters, and participants assigned. Sources of model uncertainty are discussed in Section **Error! Reference source not found.**3.3 for those sources identified as HIGH and MEDIUM significance; those identified as LOW significance are not discussed here, but can be found in Appendix A, Table A-1. Completeness, level of detail, parameter uncertainty, and other issues that were classified as being of potentially HIGH significance are discussed in more detail in Sections **Error! Reference source not found.**3.4 through **Error! Reference source not found.**3.6. The issues that were considered to be of LOW significance are not discussed further, but can be found in Appendix A.

3.2 Technical Elements of a Fire PRA

As indicated in the table, the majority of model uncertainties identified were in the Fire Scenario Selection and Analysis (FSS) technical element. This is the technical element where fire modeling is performed. Level of detail and completeness uncertainties were identified for the majority of the technical elements and a few parameter uncertainties were identified in four technical elements.

provides a summary of the number of issues identified during the workshop by Fire PRA technical element (as defined in the Level 1 Internal Fire PRA Standard (ASME/ANS, 2009)) and by uncertainty category type.

Table 2. Fire PRA Technical Elements versus Uncertainty Category (Model, Completeness, Level of Detail, Parameter) and Workshop Technical Session Issue Importance Rank (High, Medium, Low)

Technical Element	Uncertainty Category			
	# of Model Uncertainties	# of Completeness Uncertainties	# of Level of Detail Uncertainties	# of Parameter Uncertainties
Plant Boundary Definition and Partitioning (PP)	1 medium 1 low			
Fire PRA Equipment Selection (ES)	1 low	1 medium	1 high 2 medium 1 low	

Technical Element	Uncertainty Category			
	# of Model Uncertainties	# of Completeness Uncertainties	# of Level of Detail Uncertainties	# of Parameter Uncertainties
Fire PRA Cable Selection (CS)		1 medium	1 high 2 medium	
Qualitative Screening (QLS)			1 low	
Fire PRA Plant Response Model (PRM)		1 medium	2 high 2 medium	1 low
Fire Scenario Selection and Analysis (FSS)	8 high 4 medium 3 low	1 low	5 high 1 low	1 high
Fire Ignition Frequency (IGN)				2 medium 1 low
Quantitative Screening (QNS)			1 low	
Circuit Failure Analysis (CF)		1 high		1 medium
Post-fire Human Reliability Analysis (HRA)	1 high	1 high 3 medium		1 high
Fire Risk Quantification (FQ)			3 medium	
Seismic/Fire Interactions (SF)	1 medium			
Uncertainty and Sensitivity Analyses (UNC)	Issues Identified for Each Technical Element			
New Area	1 medium			

3.3 Fire PRA Model Uncertainties

The issues identified as sources of model uncertainty are summarized in this section and are presented in relation to the Level 1 Internal Fire PRA Standard high level requirement and technical element for which they are most closely related. Identified sources of model uncertainty are discussed for seven of the technical elements listed in

and one new area which could not be categorized with a technical element.

The potential HIGH sources of model uncertainty are presented first in subsection **Error! Reference source not found.**3.3.1 through 3.3.9. The identified MEDIUM sources of model uncertainty are presented in subsections 3.3.10 through 3.3.16. The LOW sources of model uncertainty are only listed in Appendix A, Table A-1 and are not discussed in detail here.

3.3.1 HIGH – Select One or More Scenarios for the Main Control Board (MCB) Involving Damage to More than One Function

Description of Issue

MCB scenarios are often risk significant. Plant knowledge and engineering judgment are needed to develop the detailed scenarios, without analyzing all possible scenarios. Simplified modeling of control room abandonment scenarios may result in either conservatism or non-conservatism. Failure to consider detailed fire-damage which can potentially fail safe shutdown outside of the control room can result in non-conservatism. This issue is considered a model uncertainty because operator efforts to shutdown the plant from outside the control room or consideration of potential fire damage affecting safe shutdown of the plant from outside the control room are not currently modeled in the PRA.

Manifestation in PRA

Typical FPRA modeling uses a bounding failure probability for control room abandonment, which may be conservative for most scenarios.

Relevance

This is relevant to the base PRA and, therefore, potentially any application, and for existing and future reactors. Because this model uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as HIGH significance. This source of model uncertainty is related to the Fire Scenario Selection and Analysis technical element discussed in Topic 21, Appendix A, Table A-1.

Potential for R&D

No potential R&D was discussed for this issue.

3.3.2 HIGH – Fire PRA Shall Include an Analysis of Potential Fire Scenarios Leading to the Main Control Room Abandonment

Description of Issue

Simplified modeling of control room abandonment scenarios due to smoke may result in either conservatism or non-conservatism.

Manifestation in PRA

Typical FPRA modeling uses a bounding failure probability for control room abandonment, which may be conservative for most scenarios. Actual modeling of safe/alternate shutdown panels outside the Main Control Room is typically not done in FPRAs and this represents a modeling uncertainty.

Relevance

This is relevant to the base PRA and, therefore, potentially any application, and for existing and future reactors. Because this model uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as HIGH significance. This source of model uncertainty is related to the Fire Scenario Selection and Analysis technical element discussed in Topic 22, Appendix A, Table A-1.

Potential for R&D

The FPRA experts determined that the NUREG/CR-6850, Appendix L (NRC, 2005) spread and control model for fires in control rooms needs to be verified.

3.3.3 HIGH – Analyze Target Damage Times Based on the Thermal Response of the Target

Description of Issue

Not considering the thermal response of damage targets in the FPRA can result in a factor of two or more conservatism. Fire testing has shown that most cables can last for 30 minutes or more given a damaging fire. Cable damage is also assumed in the FPRA when the cable tray is ignited, which may not be the case. However, there is no method presently developed to account for this issue.

Manifestation in PRA

Without consideration of thermal response (cables are typically assumed to fail when the temperature reaches a specific value), a Fire PRA can be a factor of 2 or more conservative. Additionally, an additional 20 minutes for suppression changes the risk by more than a factor of five.

Relevance

This is relevant to the base PRA and, therefore, potentially any application, and for existing and future reactors. Because this model uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as HIGH significance. This source of model uncertainty is related to the Fire Scenario Selection and Analysis technical element discussed in Topic 23, Appendix A, Table A-1.

Potential for R&D

The fire experts suggested that R&D should be carried out to develop a method for equipment damage.

3.3.4 HIGH – Fire Growth Time is Included in the Detailed Fire Scenarios

Description of Issue

Fire growth time is often treated as a constant; 12 minutes for electrical fires, and 6 or 8 minutes for typical transient fires. However, growth time can vary, and may not be independent of the heat release rate (HRR). Finally, some fires are assumed instantaneous (oil, hydrogen, etc.).

Manifestation in PRA

Many Fire PRAs are dominated by electrical cabinet fires. Growth time of 12 minutes is likely conservative for most fires. High energy arc fires can result in instantaneous growth and are treated separately.

Relevance

This is relevant to the base PRA and, therefore, potentially any application, and for existing and future reactors. Because this model uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as HIGH significance. This source of model uncertainty is related to the Fire Scenario Selection and Analysis technical element discussed in Topic 24, Appendix A, Table A-1.

Potential for R&D

More work needs to be performed to look at the correlation between growth rate and peak heat release rate. Appendix E of NUREG/CR-6850 should be looked at and verified or updated (Appendix E establishes the HRR curves).

3.3.5 HIGH – Applied Severity Factors Should be Independent of Other Factors

Description of Issue

Severity factors are generally based on conservative estimates (i.e., failure of the 1st target). Statistical and empirical models are based on generic models, and can be uncertain. Severity factors are applied either as a result of fire modeling (minimum fire heat release rate to damage cable), or using existing empirical or statistical models.

Manifestation in PRA

Severity factors are multipliers on the fire damage to reflect different size fires and HRRs.

Relevance

This is relevant to the base PRA and, therefore, potentially any application, and for existing and future reactors. Because this model uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as HIGH significance. This source of model uncertainty is related to the Fire Scenario Selection and Analysis technical element discussed in Topic 25, Appendix A, Table A-1.

Potential for R&D

The session determined that R&D should be carried out to develop new models and verify existing models for fire modeling severity factors.

3.3.6 HIGH – Apply Fire Modeling Tools to Account for Fire Growth, Damage Criteria, and Scenario Specific Attributes within the Known Limits of Applicability

Description of Issue

Application of fire modeling tools can result in either conservatism or non-conservatism. NUREG-1824 provides guidance on the use of major fire modeling tools to various conditions and parameters. However, many of the entries are listed as "yellow" where "there [are] calculated relative differences outside the experimental and model input Uncertainty." For example, all of the listed codes are listed as "yellow" for smoke concentration, which is one of the bases for the control room evacuation analysis (NRC, 2007).

Manifestation in PRA

Fire modeling tools are used to determine the magnitude of fire growth and the magnitude of target damage.

Relevance

This is relevant to the base PRA and, therefore, potentially any application, and for existing and future reactors. Because this model uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as HIGH significance. This source of model uncertainty is related to the Fire Scenario Selection and Analysis technical element discussed in Topic 29, Appendix A, Table A-1.

Potential for R&D

No potential R&D was discussed for this issue.

3.3.7 HIGH – Type of Fire Propagation Model

Description of Issue

Both intra- and inter-model uncertainty exists. Intra-model uncertainty addresses the variability that can be obtained if different models of the same type (e.g., zone or computational fluid dynamics (CFD)) are compared. Inter-model uncertainty addresses use of different types (levels) of model, usually associated with greater and lesser degrees of refinement (e.g., more detailed modeling possible via a CFD model such as FDS vs. a zonal model such as CFAST).

Manifestation in PRA

Modeling fire growth determines the level of damage modeled in the FPRA. Manipulation of the model parameters has a high impact on the Fire PRA outcome. Therefore, the PRA analyst must utilize the tools in a correct manner.

Relevance

This is relevant to the base PRA and, therefore, potentially any application, and for existing and future reactors. Because this model uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as HIGH significance. This source of model uncertainty is related to the Fire Scenario Selection and Analysis technical element discussed in Topic 30, Appendix A, Table A-1.

Potential for R&D

No potential R&D was discussed for this issue during the session.

3.3.8 HIGH – Manual Suppression

Description of Issue

Manual suppression can be applied to every fire scenario, but its model uncertainty is unknown and potentially important. Long durations, which contribute to large non-suppression probabilities, can arise from responses by extinguishers alone. This runs counter to an easy assignment of importance with respect to model uncertainty. The fire database effort underway will provide the ability to better evaluate this issue.

Manifestation in PRA

The probability of manual suppression is multiplied by the scenario frequency based on the time of the fire growth needed to cause the scenario. The manual suppression probability can include credit for the first responder and the fire brigade. The ability to adjust the manual suppression credit to reflect the fire brigade separately from the first responder is limited given current methods. For example, a general model uncertainty issue is adjusting manual suppression for fire brigade response time. Currently, the entire manual suppression curve is adjusted. Also, need to evaluate potential for Fire Brigade actions leading to additional failures.

Relevance

This is relevant to the base PRA and, therefore, potentially any application, and for existing and future reactors. Because this model uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as HIGH significance. This source of model uncertainty is related to the Fire Scenario Selection and Analysis technical element discussed in Topic 33, Appendix A, Table A-1.

Potential for R&D

The fire experts suggested that R&D should be carried out to develop a method to account for adverse actions resulting from manual suppression efforts.

3.3.9 HIGH – Determine the Time Available and Time to Perform in Support of Detailed Human Reliability Analysis (HRA)

Description of Issue

Time-lines for human error probabilities (HEPs) have uncertainty both on the time window for available time, based typically on thermal-hydraulic (T-H) analysis and the time to perform, based on simulator runs, walkdowns or talkthoughs. Fire HEPs add additional complexity, since the actions are typically in response to fire damage, which is typically conservatively estimated.

Timelines for Fire HEPs are often times based on conservative estimates for fire-damage, and best estimate but uncertainty time windows for available versus performance times.

Manifestation in PRA

Human failure events (HFE) are included in FPRA models as contributors to accident sequences. The probabilities for each HFE are evaluated based on many factors including the time to diagnose the need to perform an action and the time to actually perform the action.

Relevance

This is relevant to the base PRA and, therefore, potentially any application, and for existing and future reactors. Because this model uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as HIGH significance. This source of model uncertainty is related to the Post-Fire Human Reliability Analysis technical element discussed in Topic 49, Appendix A, Table A-1.

Potential for R&D

The session determined that R&D should be carried out to develop better guidance for evaluating timelines.

3.3.10 MEDIUM - Credited Partitioning Elements and Fire Barrier Effectiveness

Description of Issue

Uncertainty in the fire barrier failure rates should be included in multi-compartment analysis of the FPRA. The fire barrier failure rate uncertainty arises with the method of how the fire barrier penetration area is blended when performing a multi-compartment analysis.

Manifestation in the PRA

Fire barriers are utilized to delineate the fire areas into physical analysis units, and those fire barriers have failure rates which are included in the FPRA.

Relevance

This issue is relevant for the analysis of existing reactors, and it’s applicability to advanced reactors was not discussed; however, it is likely to be applicable. Because this model uncertainty has a small to moderate impact on the conclusions and risk insights, it was categorized as MEDIUM significance. This source of model uncertainty is related to the Plant Boundary Definition and Partitioning technical element discussing in Topic 2, Appendix A, Table A-1.

Potential for R&D

The regulator or industry needs to provide guidance or clarification on methods to account for multiple penetrations in a barrier.

3.3.11 MEDIUM – Establish and Apply Damage Criteria

Description of Issue

Damage criteria are developed for generic types of cable or equipment, and may vary depending on the specific cables or equipment installed. Variation within groups of cables is not large in comparison to variation among groups (e.g., thermoset versus thermoplastic).

Manifestation in PRA

Thermal damage criteria are used to determine when cables fail during a fire. Besides temperature, other affects, such as cable loading, aging, installation of metal covers, and installation specific factors can impact the time to damage for a specific cable.

Relevance

This is relevant to the base PRA and, therefore, potentially any application, and for existing and future reactors. Because this model uncertainty has a small to moderate impact on the conclusions and risk insights, it was categorized as MEDIUM significance. This source of model uncertainty is related to the Fire Scenario Selection and Analysis technical element discussed in Topic 26, Appendix A, Table A-1.

Potential for R&D

No potential R&D was discussed for this issue.

3.3.12 MEDIUM – Include Fire Growth Resulting in Propagation Between Vertical Cabinets

Description of Issue

Uncertainty exists in what is the appropriate model of fire growth resulting from propagation from cabinet-to-cabinet. NUREG/CR-6850 (NRC, 2005) includes deterministic rules on the timing and spread of fires within cabinet groups such as motor-control centers (MCCs). The recent General Electric-Hitachi report shows fire growth between cabinets is unlikely.

Manifestation in PRA

Cabinet-to-cabinet fire growth is typically important due to the potential high HRR that results (not direct equipment damage). The resulting large fire can be significant. However, most fires do not have to spread in order to damage and ignite overhead cables.

Relevance

This is relevant to the base PRA and, therefore, potentially any application, and for existing and future reactors. Because this model uncertainty has a small to moderate impact on the conclusions and risk insights, it was categorized as MEDIUM significance. This source of model uncertainty is related to the Fire Scenario Selection and Analysis technical element discussed in Topic 31, Appendix A, Table A-1.

Potential for R&D

The session suggested the need for better data to improve the models for timing and spread of fires within cabinet groups.

3.3.13 MEDIUM – Fire PRA Shall Analyze Scenarios with the Potential for Causing Fire-Induced Failure of Exposed Structural Steel

Description of Issue

Current practice in FPRA is to assume that large fires in buildings with exposed structural steel will cause building collapse. This approach may be overly conservative.

Manifestation in PRA

Scenarios are typically analyzed only when there is exposed structural steel and a high hazard source located nearby. Plant risk for damage to exposed structural steel is generally low, except for selected plants (there have been occurrences in nuclear power plant fires).

Relevance

This is relevant to the base PRA and, therefore, potentially any application, and for existing and future reactors. Because this model uncertainty has a small to moderate impact on the conclusions and risk insights, it was categorized as MEDIUM significance. This source of model uncertainty is related to the Fire Scenario Selection and Analysis technical element discussed in Topic 35, Appendix A, Table A-1.

Potential for R&D

No potential R&D was discussed for this issue.

3.3.14 MEDIUM – Fire PRA Shall Evaluate the Risk Contribution of Multi-Compartment Fire Scenarios

Description of Issue

Initially, multi-compartment analysis (MCA) was considered low risk. However, some FPRAs are showing MCA scenarios in the risk-significant scenario list. Two factors appear to impact these results:

1) Fire barrier penetration failures in NUREG/CR-6850 are uncertain, and do not clearly state if this is for a single penetration or all penetrations on an existing barrier. Additionally, treatment of barrier failure given the fire source impact is not clear.

2) Conservative fire modeling for a single area results in conservative MCA results.

Manifestation in PRA

MCA is utilized in FPRA to evaluate the potential for fire growth from one compartment to another. In performing an MCA, it is generally assumed that the combustible loading is spread throughout the fire compartment. Concentration of combustible material against a barrier could challenge the barrier.

Relevance

This source of model uncertainty is related to the Fire Scenario Selection and Analysis technical element discussed in Topic 36, Appendix A, Table A-1.

Potential for R&D

No potential R&D was discussed for this issue.

3.3.15 MEDIUM – Qualitatively Asses the Potential for Seismic/Fire Interaction Issues

Description of Issue

Seismic/fire interactions are typically only qualitatively evaluated in a FPRA. For some plants, a qualitative evaluation may miss vulnerabilities that are potentially significant.

Manifestation in PRA

The seismic/fire assessment looks at the impact of a seismic event on ignition sources, suppression and detection, plant response including brigade response, etc. The issue is treated qualitatively because it is considered low risk in relation to seismic or fire risk analyzed independently.

Relevance

This is relevant to the base PRA and, therefore, potentially any application, and for existing and future reactors. Because this model uncertainty has a small to moderate impact on the conclusions and risk insights, it was categorized as MEDIUM significance. This source of model uncertainty is related to the Seismic/Fire Interactions technical element discussed in Topic 58, Appendix A, Table A-1.

Potential for R&D

No potential R&D was discussed for this issue during the session.

3.3.16 MEDIUM – Potential for Other Hazards/Fire Interaction Issues in the PRA

Description of Issue

The potential for multiple hazards (e.g., turbine blade ejection leading to fire and flooding) occurring should be investigated as is done with seismic-fire interactions. The issue probably could be treated qualitatively due to the estimation it is considered low risk in relation to hazards occurring independently. A qualitative evaluation of other hazards/fire or fire/other hazards interaction may miss vulnerabilities that are potentially significant.

Manifestation in PRA

The session consensus was that any interaction between other hazards and fire should be analyzed in the hazard that causes the interaction.

Relevance

This is relevant to the base PRA and, therefore, potentially any application, and for existing and future reactors. Because this model uncertainty has a small to moderate impact on the conclusions and risk insights, it was categorized as MEDIUM significance. This source of model uncertainty is unrelated to any current technical element in the PRA Standard and is discussed in Topic 59, Appendix A, Table A-1.

Potential for R&D

The session determined that further guidance or method development was necessary from industry or NRC.

3.4 Fire PRA Completeness Uncertainty

Several completeness issues were identified in a number of the Fire PRA technical elements. Only the potential HIGH completeness issues are discussed here in subsections 3.4.1 and 3.4.2. The LOW and MEDIUM completeness uncertainties are presented in Appendix A, Table A-1.

3.4.1 HIGH – Apply Circuit Failure Probabilities for Undesired Spurious Operations

Description of Issue

NUREG/CR-6850 (NRC, 2005) and other Fire PRA methods do not include the probability or approach for considering spurious operation duration for DC circuits. This would include duration of spurious pilot-operated relief valve (PORV), main steam isolation valve (MSIV), and safety relief valve (SRV) openings. For some scenarios, assuming that hot shorts have a finite duration could significantly impact risk estimates.

Manifestation in PRA

For plants where MSOs contribute greatly to the overall risk, DC components typically are the most important. Short spurious operation duration will mean the component will return to its fail-safe position.

Relevance

This is relevant to the base PRA and, therefore, potentially any application, and for existing and future reactors. Because this model uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as HIGH significance. This source of model uncertainty is related to the Circuit Failure Analysis technical element discussed in Topic 48, Appendix A, Table A-1.

Potential for R&D

No potential R&D was discussed for this issue during the session.

3.4.2 HIGH – Include Operator Recovery Actions that can Restore Function

Description of Issue

Estimates for detailed Fire HEPs are highly uncertain. In addition to having high uncertainty for any recovery actions, FPRAs do not always credit recovery actions including procedural actions in the fire emergency procedures. Failure to include recovery values results in conservatism in the FPRA.

Manifestation in PRA

The addition of recovery actions is typically performed at the end of the FPRA. The total number of FPRA sequences makes the application of recovery actions difficult.

Relevance

This is relevant to the base PRA and, therefore, potentially any application, and for existing and future reactors. Because this level of detail uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as HIGH significance. This level of detail issue is related to the Post-Fire Human Reliability Analysis technical element discussed in Topic 54, Appendix A, Table A-1.

Potential for R&D

No potential R&D was discussed for this issue during the session.

3.5 Fire PRA Level of Detail Issues

Several level of detail issues were identified in a number of the Fire PRA technical elements. Only the potential HIGH level of detail issues are discussed here in subsections 3.5.2 through 3.5.9. The LOW and MEDIUM level of detail issues are presented in Appendix A, Table A-1.

3.5.1 HIGH – Develop One or More Fire Scenarios for Each Unscreened Area Such that Risk is Characterized or Bounded

Description of Issue

FPRAs performed using NUREG/CR-6850 involve development and analysis of thousands of scenarios (NRC, 2005). Typically, most are conservatively modeled. For higher risk areas, it is possible to develop detailed scenarios where risk-relevant scenarios are not fully developed. Therefore, this issue is classified as level of detail because further refined modeling of fire scenarios can be performed. The central issue raised questions the level of application of the available models.

Manifestation in PRA

A typical FPRA includes both:

- Scenarios where the ignition source and target grouping are conservatively modeled, resulting in conservative risk results.
- Incomplete scenario development where not all risk-relevant combinations of ignition sources and targets are developed.

Relevance

This is relevant to the base FPRA and, therefore, potentially any application, and for existing and future reactors. Because this level of detail uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as having a HIGH significance. This level of detail issue is related to the Fire Scenario Selection and Analysis technical element discussed in Topic 20, Appendix A, Table A-1.

Potential for R&D

No potential R&D was discussed for this issue.

3.5.2 HIGH – Equipment is Selected that May Cause a Failure of a Safe Shutdown Component, Including Spurious Operation

Description of Issue

Incomplete equipment selection in the FPRA model may result in an under prediction of risk. However, if equipment not selected is assumed failed in the FPRA, then the risk results will be conservative.

Manifestation in PRA

Fire PRAs include additional equipment not included in an internal event PRA. The additional equipment is added to reflect potential failure modes induced by fire-induced failure of cables and include such failure modes as spurious opening or closing of valves, spurious start of pumps, and spurious instrument readings.

Relevance

This is relevant to the base PRA and, therefore, potentially any application, and for existing and future reactors. Because this level of detail uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as HIGH significance. This level of detail issue is related to the Fire PRA Equipment Selection technical element discussed in Topic 6, Appendix A, Table A-1.

Potential for R&D

No potential R&D was discussed for this issue during the session.

3.5.3 HIGH - Cables and Circuits Impacting Selected Equipment are Identified and Traced

Description of Issue

Assumed cable routing is typically performed for credited non-safety components, such as Main Feedwater and Condensate. Assumed Cable Routing may be inaccurate and may result in under or over prediction of risk, depending on the fire area.

Manifestation in PRA

The location of control and power cables should be determined to the extent possible in order to determine the impact of fires on essential equipment. The effort to trace cables can be difficult and in some cases, the location of cables may have to be assumed.

Relevance

This is relevant to the base PRA and, therefore, potentially any application, and for existing and future reactors. Because this level of detail uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as HIGH significance. This level of detail issue is related to the Fire PRA Cable Selection technical element discussed in Topic 10, Appendix A, Table A-1.

Potential for R&D

No potential R&D was discussed for this issue during the session.

3.5.4 HIGH - The Fire Plant Response Model Includes the Fire-Induced Initiating Events or Accident Sequences

Description of Issue

Over-simplification or failure to model new initiating events or accident sequences can result in an under-prediction of risk. Given numerous new scenarios added as a result of MSOs, failure to model these accurately can result in significant errors.

Manifestation in PRA

The FPRA model utilizes the initiating events and accident scenarios from the internal events PRA. Fire-induced failures are incorporated into the internal event models as appropriate.

Relevance

This is relevant to the base PRA and, therefore, potentially any application, and for existing and future reactors. Because this level of detail uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as HIGH significance. This level of detail issue is related to the Fire PRA Plant Response Model technical element discussed in Topic 15, Appendix A, Table A-1.

Potential for R&D

No potential R&D was discussed for this issue during the session.

3.5.5 HIGH – Modify the Plant Response Model to Include New Equipment, Including Spurious Operations

Description of Issue

Incorrect modeling or failure to model new equipment may result in an underestimate of risk. Typically there are a large number of modeling changes to support the FPRA. It is common that the modeling is complicated, involving logic specific for the location of the fire.

Manifestation in PRA

The FPRA model modifies the internal events PRA to reflect fire-induced failures. Fire-induced failures are incorporated into the internal event models as appropriate.

Relevance

This is relevant to the base PRA and, therefore, potentially any application, and for existing and future reactors. Because this level of detail uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as HIGH significance. This level of detail issue is related to the Fire PRA Plant Response Model technical element discussed in Topic 17, Appendix A, Table A-1.

Potential for R&D

No potential R&D was discussed for this issue during the session.

3.5.6 HIGH – Assume Damaged Cable if Exact Cable Routing is Unknown

Description of Issue

It is not uncommon to not know specifically in a room where every cable is located. As a result, the FPRA assumes the cable is damaged for every fire until the cable is traced in detail. This has shown up as a major conservatism in several FPRAs.

Manifestation in PRA

The location of cables is used in a FPRA to determine the equipment that can be damaged by a fire and the specific failure mode of the component.

Relevance

This is relevant to the base PRA and, therefore, potentially any application, and for existing and future reactors. Because this level of detail uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as HIGH significance. This level of detail issue is related to the Fire Scenario Selection and Analysis technical element discussed in Topic 37, Appendix A, Table A-1.

Potential for R&D

No potential R&D was discussed for this issue during the session.

3.5.7 HIGH – Selection of One or More Scenarios for the Main Control Board Involving Damage to More than One Function

Description of Issue

Main control board (MCB) scenarios are often risk significant. Level of detail issues arise with incomplete scenario development where not all risk-relevant combinations of ignition sources and targets are developed. Plant knowledge and engineering judgment are needed to develop the detailed scenarios, without analyzing all possible scenarios.

Manifestation in PRA

Fires in the control room are analyzed in a FPRA. Fires within particular control boards are generally considered.

Relevance

This is relevant to the base PRA and, therefore, potentially any application, and for existing and future reactors. Because this level of detail uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as HIGH significance. This level of detail issue is related to the Fire Scenario Selection and Analysis technical element discussed in Topic 38, Appendix A, Table A-1.

Potential for R&D

No potential R&D was discussed for this issue during the session.

3.5.8 HIGH – Characterize Factors that will Influence the Timing and Extent of Fire Damage for Each Combination of Ignition Source and Damage Target Sets

Description of Issue

Realistic fire modeling for each scenario requires a significant effort. Typically, a majority of the scenarios are conservatively modeled. Fire damage estimates typically start conservative, with more realism added to the top (risk-significant) scenarios. Details may include multiple heat release rate groups, inclusion of fire growth time, decay time, consideration for environmental conditions for realistic time to damage, and more detailed configuration considerations.

Manifestation in PRA

Detailed fire modeling is time-consuming and is only performed for a limited set of significant scenarios. This level of detail issue could result an over prediction of risk.

Relevance

This is relevant to the base PRA and, therefore, potentially any application, and for existing and future reactors. Because this level of detail uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as HIGH significance. This level of detail issue is related to the Fire Scenario Selection and Analysis technical element discussed in Topic 39 Appendix A, Table A-1.

Potential for R&D

No potential R&D was discussed for this issue during the session.

3.5.9 HIGH –Assessment of Fire Suppression Effectiveness for Each Fire Scenario Being Analyzed

Description of Issue

Credit for fire suppression is typically performed once the time to damage is determined from fire modeling. Generally speaking, with detailed fire modeling only performed on a small percentage of scenarios, the credit for suppression is conservative. Additionally, the existing NUREG/CR-6850 suppression curves are considered conservative (no credit for control of fires prior to suppression) (NRC, 2005). Estimates of conservatism for suppression are a factor of two for a typical Fire PRA.

Manifestation in PRA

In FPRAs, a time to suppression curve is used to determine the probability of fire suppression before damage occurs.

Relevance

This is relevant to the base PRA and, therefore, potentially any application, and for existing and future reactors. Because this level of detail uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as HIGH significance. This level of detail issue is related to the Fire Scenario Selection and Analysis technical element discussed in Topic 41, Appendix A, Table A-1.

Potential for R&D

No potential R&D was discussed for this issue during the session.

3.6 Fire PRA Parameter Uncertainties

Several parameter uncertainty issues were identified in a number of the Fire PRA technical elements. Only the potential HIGH parameter uncertainty issues are discussed here in subsections 3.6.1 and 3.6.2. The LOW and MEDIUM level of detail issues are presented in Appendix A, Table A-1.

3.6.1 HIGH – Estimate Fire Modeling Parameters Based on Relevant Generic Industry and Plant-Specific Information

Description of Issue

Fire modeling parameter estimates are typically either well known or applied as bounding estimates. This may include factors like room temperature, heating, ventilation and air conditioning (HVAC) flow, wall material and thickness, etc.

Manifestation in PRA

Several parameters are utilized in the modeling of fire growth utilized in a FPRA. These parameters can affect the equipment that can be damaged from different fire scenarios.

Relevance

This is relevant to the base PRA and, therefore, potentially any application, and for existing and future reactors. Because this model uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as HIGH significance. This source of model uncertainty is related to the Fire Scenario Selection and Analysis technical element discussed in Topic 34, Appendix A, Table A-1.

Potential for R&D

No potential R&D was discussed for this issue during the session.

3.6.2 HIGH – Perform Detailed Human Error Probability (HEP) Analysis for Significant HEPs, Including Performance Shaping Factors (PSFs) from Fire

Description of Issue

Results of detailed HEP analysis, especially when considering the fire-specific PSFs, is highly uncertain. Generally, most HEPs are lower risk. However, a few key HEPs are typically in the dominant sequences, such as control room evacuation. Estimates for detailed Fire HEPs are highly uncertain.

Manifestation in PRA

The addition of recovery actions is typically performed at the end of the FPRA. The total number of FPRA sequences makes the application of recovery actions difficult.

Relevance

This is relevant to the base PRA and, therefore, potentially any application, and for existing and future reactors. Because this level of detail uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as HIGH significance. This level of detail issue is related to the Post-Fire Human Reliability Analysis technical element discussed in Topic 53, Appendix A, Table A-1.

Potential for R&D

No potential R&D was discussed for this issue during the session.

4. UNCERTAINTIES IDENTIFIED FOR SEISMIC EVENTS PRA

4.1 Introduction

During the morning Seismic PRA session of the NRC/EPRI Workshop on PRA Uncertainties, four experts from both industry and the NRC made presentations on the sources of uncertainty in a seismic PRA. The expert presenters followed differing formats, which complicated assimilating the viewpoints on different aspects of the PRA uncertainties. One presenter discussed a few particular uncertainty examples for each of the three seismic PRA elements. Another presenter focused principally on the uncertainties related to probabilistic seismic hazard analysis and presented a detailed discussion on this element. A third presenter focused mainly on the seismic fragility evaluation and discussed the nuances and associated uncertainties involved in that element. The fourth presenter illustrated some seismic PRA uncertainties by presenting the results of two recent seismic PRAs.

In the afternoon session, far ranging discussions on seismic PRA were carried out by many of the session attendees along with the expert presenters of the morning. Finally, the session participants stepped through the various uncertainties raised in the discussions in the order of the three technical elements listed in the ASME/PRA Standard for seismic PRA: probabilistic seismic hazard analysis, seismic fragility evaluation, and seismic plant response analysis. Uncertainties for each element were discussed and ranked as to their significance with the predetermined scale of high, medium or low significance. Model and parameter uncertainties, as well as level of detail issues of various rankings were identified and are discussed below.

4.2 Technical Elements of a Seismic Events PRA

The ASME/ANS PRA Standard (ASME/ANS, 2009) identifies three technical elements which are probabilistic seismic hazard analysis, seismic fragility evaluation and seismic plant response analysis. The following presents a description of these three technical elements and a synopsis of discussion amongst seismic session participants:

- **Probabilistic seismic hazard analysis** develops the seismic hazard curve considering issues such as: seismic source characterization modeling (the SSC logic tree), seismic source data, seismic source location and geometry, maximum earthquake magnitude, earthquake recurrence, hazard uncertainty characterization, ground motion characterization modeling, also referred to as the ground motion characterization (GMC) logic tree, ground motion attenuation prediction equations, effects of local site characteristics on ground motion, propagation of uncertainty, and site specific response spectral shape.

The seismic hazard analysis discussion centered on the uncertainties in characterizing the seismic sources, characterizing the ground motion attenuation models, which are now called ground motion prediction equations (GMPEs), and characterizing the response at the site of interest.

The seismic source characterization (SSC) model provides the characterization for all seismic sources that may impact a site of interest. This involves characterizing all seismic sources that could impact the site, determining every earthquake that each source can produce and the likelihood of the earthquake, assesses the ground motion distribution for each earthquake, and integrating the ground motion over all earthquakes accounting for the likelihood of each scenario. The SSC model accounts for epistemic uncertainty in the form of a logic tree composed of the full suite of alternative technically defensible interpretations of the data. The workshop participants agreed that there had been some progress in this area with the publication of the Central and Eastern U.S. Seismic Source Characterization Study for Nuclear Facilities model (generally called the CEUS SSC model), which was developed using a Senior Seismic Hazard Analysis Committee (SSHAC) Level 3 process. The study was conducted by the NRC and EPRI, as well as the Department of Energy (DOE), and was jointly published as NUREG-2115, EPRI 1021097, and DOE/NE-0140.

Regarding the GMC, the workshop participants agreed that this involved high uncertainty, and probably involved the largest uncertainty in the hazard analysis as well as the whole seismic PRA. The GMC model provides a distribution of predicted ground motions for a particular magnitude distance scenario earthquake. Again a logic tree accounts for epistemic uncertainty. The GMC model is composed of GMPEs, which have a high associated uncertainty. Workshop participants discussed the development which is underway called the NGA-East (Next Generation Attenuation Relationships for Central and Eastern North America) project, with results expected by 2014. This project is being conducted as a SSHAC Level 3 study, and is sponsored by the NRC and EPRI, as well as the US Geological Survey and DOE. This new model won't necessarily reduce the estimate of uncertainty from current values until more data on earthquakes is obtained and updated, but will provide a more robust and technically defensible estimate of the uncertainty. It will also provide a more technically defensible estimate of the median values and the shape of the hazard curve.

For both the SSC and GMC workshop participants agreed that the SSHAC process, as discussed in NUREG/CR-6372 and NUREG-2117, provides a robust approach to many of the requirements in the ASME/ANS PRA Standard. The SSHAC process also provides a way to transparently and systematically capture many of the uncertainties in the SSC and GMC characterizations.

Regarding site response, workshop participants agreed that there was high uncertainty for soil sites, and considerably lower uncertainty for rock sites. There was also discussion of lack of data for sites, with many plants lacking modern equipment for data acquisition. Another issue discussed was the lack of standardization of techniques to evaluate site response. Some participants advocated for standardized techniques across plants for site response.

- **Seismic fragility evaluation,** which is based on analyses, testing, and experience data, is developed considering such complex issues as soil-structure interaction, incoherency effect on structural response, etc., Some participants voiced the need for an effort to standardize the fragility analysis for nuclear plant structures, systems and components in a handbook, an effort believed to be able to significantly reduce both parameter and model uncertainties. This would also help to increase the number of qualified practitioners, which is currently very low.

  Workshop participants felt that the uncertainty in seismic structure capacity is of medium significance, while the uncertainty in the in-structure response it is high. Some participants felt that soil-structure-interaction modeling is not well integrated with probabilistic seismic hazard analysis for or seismic PRA in terms of carrying through probabilistic loading.

- **Seismic plant response analysis** is usually based on the internal events plant response model, but is developed with consideration of such issues as adding passive components and structures to the model while simplifying some of the details of the internal events model. All workshop participants agreed that the major uncertainty of this element was the modeling of human performance following an earthquake, which is especially challenging for strong ground motion events. Participants noted that modeling some human performance events in an earthquake event can be very important in seismic PRA, and that adequate modeling of human reliability is more important for low power and shutdown (LPSD) analyses than for at-power analyses.

Workshop participants also discussed a number of uncertainty issues that do not fit neatly into a particular seismic PRA technical element. These miscellaneous issues included a discussion on the problem caused by the rapidly changing data and techniques used in conducting a seismic PRA. While there is rapid advancement in both areas, this brings an uncertainty related question regarding the viability and quality of older seismic studies. The question is whether, for a particular application, updating an older study is worthwhile and acceptable, compared to conducting a new study. This would have to be decided on a case-by-case basis.

Workshop participants also learned of several additional current or proposed research projects, besides the NGA-East seismic hazard study mentioned above, that are attempting to address seismic PRA uncertainties. These efforts included investigating the scenario earthquake approach, the fragility of potentially high frequency sensitive components, the correlation of performance of similar components under seismic loading, and comparison of more detailed structural modeling with conventional simplified modeling.

Some workshop participants also recommended that a discussion on how to deal with decision-making should be included in NUREG-1855, in particular NUREG-1855 could use a discussion on how to disposition issues that have large uncertainty in risk-informed decision making.

The results of the Seismic PRA sessions of the NRC/EPRI Workshop on PRA Uncertainties with respect to the uncertainties identified are presented in the next sections of this chapter. The uncertainties are grouped by whether they are model uncertainties, level of detail issues, or parameter uncertainties, and are ordered first by significance and within significance by technical element. A summary of the identified uncertainties is presented in

Table 3 below.

Table 3. Seismic Events PRA Technical Elements versus Uncertainty Category (Model, Completeness, Level of Detail, Parameter) and Workshop Technical Session Issue Importance Rank (High, Medium, Low)

Technical Element	Uncertainty Category			
	# of Model Uncertainties	# of Completeness Uncertainties	# of Level of Detail Uncertainties	# of Parameter Uncertainties
Probabilistic Seismic Hazard Analysis (SHA)	2 high			2 high 1 medium
Seismic Fragility Evaluation (SFR)	2 high 1 medium 4 low		3 medium	2 medium
Seismic Plant Response Analysis (SPR)	1 high 2 medium		1 medium	
Other	1 high			

It is very important to note that in the table above and in the subsequent sections, the type of uncertainty is portrayed by the manner in which it is manifested in the PRA model itself, not in the supporting analyses. For example, while most of the uncertainties in the probabilistic seismic hazard development that are related to seismic source characterization as well as ground motion attenuation characterization can be traced back to the various models used in the characterization, if the way the uncertainty shows up in the PRA is a distribution of a parameter range, then the uncertainty is portrayed as a parameter uncertainty.

4.3 Seismic Events PRA Model Uncertainties

The issues identified as sources of model uncertainty are summarized in this section and are presented in relation to the Level 1 Seismic Events PRA Standard high level requirement and technical element for which they are most closely related (ASME/ANS, 2009). Identified sources of model uncertainty are discussed for nine of the technical elements listed in

Table 3.

The potential HIGH sources of model uncertainty are presented first in subsection 0 through 4.3.5. The identified MEDIUM sources of model uncertainty are presented in subsections 4.3.6 through 0. The LOW sources of model uncertainty are only listed in Appendix B, Table A-2 and are not discussed in more detail here.

4.3.1 HIGH – Site response: Simplification and Lack of Standardization

Description of Issue

Site response has significant uncertainty and a potentially large effect on the hazard results. However, site response techniques are not as standardized as they could be. Simplifying assumptions do not always apply and other tools are not well developed. Spatial and material variability is not always well captured and randomization approaches and tools are limited.

Manifestation in PRA

The choice of model used to characterize site response from limited data is a model uncertainty that can have a significant impact on the PRA results.

Relevance

This uncertainty is relevant to both the base PRA and any application involving seismic analysis, and is relevant for existing and new reactors. Since this uncertainty has a high impact on the PRA conclusions and risk insights, it was categorized as having HIGH significance. This source of uncertainty is related to the Probabilistic Seismic Hazard Analysis technical element discussed in Topic 3, Appendix B, Table A-2.

Potential for R&D

No new potential R&D was discussed for this issue during the session. However, the NRC has ongoing research in this area.

4.3.2 HIGH – Spectral Shape

Description of Issue

Different approaches to developing the spectral input lead to different answers. Uncertainty in spectral shape arises from both the GMPEs and the use of a scenario earthquake or uniform hazard response spectra. The use of uniform hazard is usually conservative for design and for use in seismic PRA.

Manifestation in PRA

Spectral shapes must be appropriate. They can be based on deaggregation or on a uniform hazard spectrum approach. This is a model uncertainty that can have a significant impact on the results.

Relevance

This is relevant to the base PRA and, therefore, potentially any application, and for existing and future reactors. Because this model uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as having HIGH significance. This source of model uncertainty is related to the Probabilistic Seismic Hazard Analysis technical element discussed in Topic 5, Appendix B, Table A-2.

Potential for R&D

No potential R&D was discussed for this issue during the session.

4.3.3 HIGH - Soil Structure Interaction (SSI)

Description of Issue

Soil-structure interaction is very site specific. Soil-structure-interaction modeling is not well integrated with seismic hazard analysis or seismic PRA in terms of carrying through probabilistic loading.

Manifestation in PRA

The modeling of the soil-structure interaction will significantly influence the hazard demand on the systems, structures and components modeled in the plant PRA model. This is a modeling uncertainty.

Relevance

This uncertainty is relevant to both the base PRA and any application involving seismic analysis, and is relevant for existing and new reactors. This model uncertainty was categorized as having HIGH significance, especially for soil sites. This source of uncertainty is related to the Seismic Fragility Evaluation technical element discussed in Topic 6, Appendix B, Table A-2.

Potential for R&D

No potential R&D was discussed for this issue during the session.

4.3.4 HIGH – Treatment of Human Error under Seismic Conditions

Description of Issue

The approach used for treating human error under seismic conditions is relatively crude. Human factors are not well characterized and may be very site specific

Manifestation in PRA

The human reliability analysis (HRA) models used in PRA can have significant influence on the results. A few actions can have a large impact in a seismic PRA. This is a modeling uncertainty in the PRA.

Relevance

This uncertainty is relevant to both the base PRA and any application involving seismic analysis, and is relevant for existing and new reactors. This model uncertainty was categorized as having HIGH significance. This source of uncertainty is related to the Seismic Plant Response Analysis technical element discussed in Topic 18, Appendix B, Table A-2.

Potential for R&D

Improved human failure rate modeling for seismic conditions should be pursued. Suggestions included adapting fire HRA model methods with different stresses, using performance shaping factors that are used to analyze HRA in context of the scenario.

4.3.5 HIGH – Seismic PRA Updating

Description of Issue

Knowledge regarding seismic data and analysis techniques has evolved rapidly and significantly. There is uncertainty about the quality or viability of older seismic studies and the role of engineering judgment used. This can be an issue when a new analysis is used to update an old study, rather than to replace it.

Manifestation in PRA

Specific guidance (based on guidance in the ANSE/ANS standard) is provided for situations in which an update should performed. However, the quality of the technical basis of an older study is often a subjective decision. This modeling uncertainty can affect the entire basis of the seismic analysis and lead to a big difference in results.

Relevance

This issue comes under modeling uncertainty and is relevant to both the base PRA and any application involving seismic analysis, and is relevant for existing and new reactors. This model uncertainty was categorized as having HIGH significance for many cases. This source of uncertainty is related to all the technical elements and discussed in Topic 22, Appendix B, Table A-2.

Potential for R&D

No potential R&D was discussed for this issue during the session.

4.3.6 MEDIUM – Functional Failure Modes Not Clearly Tied to the Structural Deformations

Description of Issue

The relationship between the structure drift resulting from the seismic variable being used to describe the seismic hazard and the functional failure of the equipment attached to the structure is at best nebulous.

Manifestation in PRA

This uncertainty manifests as a model uncertainty in the PRA. Assumptions regarding the functional failure of the systems, structures and components relative to the seismic motion of the structure can significantly influence the PRA results.

Relevance

This uncertainty is relevant to both the base PRA and any application involving seismic analysis, and is relevant for existing and new reactors. This model uncertainty was categorized as having MEDIUM significance. This source of uncertainty is related to the Seismic Fragility Evaluation technical element discussed in Topic 7, Appendix B, Table A-2.

Potential for R&D

No potential R&D was discussed for this issue during the session.

4.3.7 MEDIUM-LOW – Generic Conversion of HCLPF to Fragility

Description of Issue

In some seismic PRA applications, the so-called hybrid method is used wherein the high confidence low probability of failure (HCLPF) capacity is calculated using the conservative deterministic failure margin (CDFM) method and the median capacity is estimated using a generic βc value. Using these parameters, the mean capacity and hence the mean fragility curve are approximated.

Manifestation in PRA

This approximate method of obtaining fragility manifests itself as a modeling uncertainty in the PRA since it influences not only the fragility parameter range but also the shape of the mean fragility curve.

Relevance

This uncertainty is relevant to both the base PRA and any application involving seismic analysis, and is relevant for existing and new reactors. This model uncertainty was categorized as having MEDIUM to LOW significance. This source of uncertainty is related to the Seismic Fragility Evaluation technical element discussed in Topic 8, Appendix B, Table A-2.

Potential for R&D

No potential R&D was discussed for this issue during the session.

4.3.8 MEDIUM-LOW – Treatment of Correlation

Description of Issue

The treatment of correlation is usually "one fails-all fails," since the approach often taken in seismic PRA is to assume 100% response correlation as a starting point. If the issue of correlation then seems to make a difference to the overall results or insights, one can do a sensitivity analysis by assuming zero response correlation to ascertain how important the correlation might be, but sensitivity studies are often not thoroughly performed.

Manifestation in PRA

This is a modeling uncertainty that usually makes a difference for a few components (like diesel generators) but for most cases it does not lead to a big difference in results. However, it can be essential for some applications.

Relevance

This uncertainty is relevant to both the base PRA and any application involving seismic analysis, and is relevant for existing and new reactors. This model uncertainty was categorized as having LOW significance for many cases but can be MEDIUM for some applications. This source of uncertainty is related to the Seismic Plant Response Analysis technical element discussed in Topic 19, Appendix B, Table A-2.

Potential for R&D

The NRC (with participation from EPRI) is currently carrying out a project to address this topic.

4.3.9 MEDIUM-UNKNOWN – Seismically-Induced Fire and Flooding are Not Well Developed or Integrated

Description of Issue

Seismic induced fire and flood are usually treated in a qualitative manner in a seismic analysis. These items are disposed of usually via qualitative evaluation during walkdowns (and for floods, review of dams and ponds near the site); some quantitative studies have been performed.

Manifestation in PRA

Seismic induced fire and flood are usually treated in a qualitative manner in a seismic analysis and their significance is unknown. This is a modeling uncertainty in the PRA.

Relevance

This issue comes under modeling uncertainty and is relevant to both the base PRA and any application involving seismic analysis, and is relevant for existing and new reactors. This model uncertainty was categorized as having UNKNOWN significance and was assigned a MEDIUM value by default. This source of uncertainty is related to the Seismic Plant Response Analysis technical element discussed in Topic 20, Appendix B, Table A-2.

Potential for R&D

No potential R&D was discussed for this issue during the session.

4.4 Seismic Events PRA Completeness Uncertainty

No seismic uncertainties were identified as completeness uncertainties. However, as noted in Section 2.1.2 above, completeness uncertainty also can be thought of as a type of model uncertainty. For example, the seismic model uncertainty discussed in 4.3.9, "Seismically-Induced Fire and Flooding are Not Well Developed or Integrated," could be considered a completeness uncertainty. Likewise, some of the level of detail uncertainties discussed below, such as 4.5.1 "Conservative Assumption of Structural Failures," 4.5.2 "Use of Surrogate Elements," and 4.5.4 "Simplification of the System Model" could be considered under completeness uncertainty.

4.5 Seismic Events PRA Level of Detail Issues

Several level of detail issues were identified in a number of the Seismic PRA technical elements. Unlike the other uncertainty workshop sessions, the seismic session only identified MEDIUM significance level of detail issues. Therefore, where the other sections were limited to discussing those topics ranked as HIGH, the seismic section presented here addresses only MEDIUM. The MEDIUM level of detail uncertainties are discussed in subsections 4.5.1 through 4.5.4

4.5.1 MEDIUM – Conservative Assumption of Structural Failures

Description of Issue

In the conduct of seismic PRAs, usually conservative assumptions are made regarding structural failures of structures and components. This is done to make the analysis more efficient. For example, for the sake of efficiency the structure, system and component (SSC) is considered failed with the onset of yielding or buckling. Actually, the SSC may be able to carry out its function beyond the point of yield or buckling.

Manifestation in PRA

This uncertainty manifests as a level of detail issues in the PRA. Conservative assumptions regarding structural failure may bias the PRA results, and may mask contributions of fragility of one SSC with regard to another.

Relevance

This uncertainty is relevant to both the base PRA and any application involving seismic analysis, and is relevant for existing and new reactors. Since this level of detail was judged to have a moderate impact on the PRA conclusions and risk insights, it was categorized as having MEDIUM significance. This source of uncertainty is related to the Seismic Fragility Evaluation technical element discussed in Topic 9, Appendix B, Table A-2.

Potential for R&D

This is really a level of detail issue and the uncertainty could be reduced with more detailed analyses of failure modes, but such an effort is likely to be quite costly.

4.5.2 MEDIUM - Use of Surrogate Elements

Description of Issue

Attempts to capture the risk contribution via "surrogate" elements in seismic PRAs have not been very successful in the past. The ASME/ANS PRA Standard does not recommend their use. Analysts have rarely redone the core damage frequency calculations for different screening levels to assess the completeness issue.

Manifestation in PRA

This uncertainty manifests as a completeness/level of detail issue in the PRA. The use of surrogate elements may influence the PRA results, and mask potentially significant contributions of one or more systems, structures or components embedded in the surrogate element.

Relevance

This uncertainty is relevant to both the base PRA and any application involving seismic analysis, and is relevant for existing and new reactors. This completeness/level of detail issue was categorized as having MEDIUM significance. This source of uncertainty is related to the Seismic Fragility Evaluation technical element discussed in Topic 10, Appendix B, Table A-2.

Potential for R&D

No potential R&D was discussed for this issue during the session, but the obvious implication is not to use surrogate elements but rather a more detailed model (which will likely lead to a higher cost PRA).

4.5.3 MEDIUM - Structure Modeling

Description of Issue

This issue was only briefly mentioned in the workshop but concerns the level of detail at which structures in a seismic PRA are modeled, for example in a simplified “stick” model or a more detailed finite element model.

Manifestation in PRA

This uncertainty manifests as a level of detail issue in the PRA. The use of structure modeling detail will influence the PRA results.

Relevance

This uncertainty is relevant to both the base PRA and any application involving seismic analysis, and is relevant for existing and new reactors. This level of detail issue was categorized as having MEDIUM significance. This source of uncertainty is related to the Seismic Fragility Evaluation technical element discussed in Topic 11, Appendix B, Table A-2.

Potential for R&D

No potential R&D was discussed for this issue during the session, but this issue was mentioned as being one that current or proposed research projects are attempting to address.

4.5.4 MEDIUM-LOW – Simplification of the System Model

Description of Issue

Since many passive components and structures have to be included in a seismic PRA, for the sake of efficiency the seismic PRA plant response model usually starts with an internal events model that is simplified via various assumptions on initiating events and systems, structures and components (SSCs). This results in a simplified system model with a limited number of SSCs.

Manifestation in PRA

This uncertainty manifests as a completeness/level of detail issue in the PRA. The simplified model may influence the PRA results, and miss potentially significant contributions of one or more SSCs not modeled due to the simplification.

Relevance

This uncertainty is relevant to both the base PRA and any application involving seismic analysis, and is relevant for existing and new reactors. This completeness/level of detail issue was categorized as having MEDIUM significance. This source of uncertainty is related to the Seismic Plant Response technical element discussed in Topic 21, Appendix B, Table A-2.

Potential for R&D

No potential R&D was discussed for this issue during the session, but a more detailed model (which will likely lead to a higher cost PRA) would address this uncertainty.

4.6 Seismic Events PRA Parameter Uncertainties

Several parameter uncertainty issues were identified in a number of the Seismic PRA technical elements. Only the potential HIGH parameter uncertainty issues are discussed here in subsections 4.6.1 and 0.

4.6.1 HIGH – Ground Motion Characterization (GMC)

Description of Issue

Ground Motion Prediction Equations (GMPEs) provide a distribution of predicted ground motions for a particular magnitude and distance scenario earthquake. A Ground Motion Characterization (GMC) model incorporates a suite of appropriate and technically defensible Ground Motion Prediction Equation (GMPEs) into a logic tree framework for use in a Probabilistic Seismic Hazard Analysis (PSHA) conducted for a particular site. A host of specific technical questions related to development of GMPE models are a matter of current expert discussion. A new suite of GMPE models for use in central and eastern North America, along with guidance as to how they are to be combined to form GMC models, are being developed in the NGA-East project under development. The current models have been generally hampered by the lack of data available at the time of their development. There is currently uncertainty resulting from both the available data (or lack thereof) and the appropriate GMPEs to use within the GMC model. The workshop group did not get into the specific technical questions of GMPE and GMC model development but understood that the uncertainty in the GMC models tend to drive the uncertainty in PSHA analyses.

Manifestation in PRA

The GMPE models used to characterize ground motion attenuation, based on limited data, have many uncertainties, but ultimately it is the output of these models that is used to determine an appropriate range of hazard parameters for the ground motion characterization in the PRA model. So in the PRA this uncertainty manifests itself as a parameter uncertainty.

Relevance

This uncertainty is relevant to both the base PRA and any application involving seismic analysis, and is relevant for existing and new reactors. Since this uncertainty has a high impact on the PRA conclusions and risk insights, it was categorized as having HIGH significance. This source of uncertainty is related to the Probabilistic Seismic Hazard Analysis technical element discussed in Topic 2, Appendix B, Table A-2.

Potential for R&D

The workshop group noted that this uncertainty is captured in the PRA model by the SSHAC (Senior Seismic Hazard Analysis Committee) process (NUREG/CR-6372 and NUREG-2117), even if large. There is a current effort underway to develop better data and GMPEs for the central and eastern US through the NGA-East project, which is being conducted using the SSHAC Level 3 process. NGA-east isa follow on to the successful NGA-West project that has significantly improved the suite of GMPE models available to western sites.

4.6.2 HIGH – Site Response: Lack of Geotechnical Information

Description of Issue

Site response has significant uncertainty and a potentially large effect on the hazard results, but many operating plants lack geotechnical information from modern equipment for their sites.

Manifestation in PRA

The models used to characterize site response from limited data have many uncertainties, but ultimately it is the output of these models that is used to determine an appropriate range of hazard parameters for the site response characterization in the PRA model. So in the PRA this uncertainty manifests itself as a parameter uncertainty.

Relevance

This uncertainty is relevant to both the base PRA and any application involving seismic analysis, and is relevant for existing and new reactors. Since this uncertainty has a high impact on the PRA conclusions and risk insights, it was categorized as having HIGH significance. This source of uncertainty is related to the Probabilistic Seismic Hazard Analysis technical element discussed in Topic 4, Appendix B, Table A-2.

Potential for R&D

Obtaining better site-specific data, as a not very expensive option for improving on this uncertainty, was discussed for this issue during the session.

5. UNCERTAINTIES IDENTIFIED FOR LOW POWER AND SHUTDOWN (LPSD) PRA

5.1 Introduction

There were many topics raised for discussion during the session on Low Power and Shutdown (LPSD) PRAs. One conclusion was that the low power plant operating states (POSs) are essentially modeled in the same way as the at-power POS, with some minor differences which is in contrast to shutdown POSs, where the plant configuration may be quite different from at-power. Therefore, the sources of uncertainty for low power POSs are largely the same, and are not repeated here. The focus therefore, was on the sources of uncertainty unique to the modeling of the shutdown POS, including those that arise in the definition of the POSs.

During the LPSD session, relatively few specific sources of model uncertainty were identified. While several issues were raised, the majority were determined, upon group discussion, not to be sources of model uncertainty because they did not satisfy the following criteria:

- A model uncertainty only arises because of a lack of knowledge about how to model some aspect of the plant response or the effect of specific failures.
- To address these sources of model uncertainty, the PRA model is constructed making specific assumptions.

The majority of the issues raised were related to approximations and the level of detail that is needed to produce a realistic assessment of the risk from LPSD operations. While these result in uncertainties in the results of the PRA, they are essentially resolvable by including more detail. A few issues were identified related to completeness in that there are some potential contributors to risk that may not be typically included in LPSD PRA models. The participants at the workshop seemed to be in general agreement that the reason for this is that there is considerably less experience with performing LPSD PRAs compared to at-power PRAs and in particular fewer peer reviews or NRC reviews, and consequently a smaller population of models.

While LPSD PRAs have been used for some applications (e.g., benchmarking defense-in-depth shutdown models, understanding shutdown risk to characterize Significance Determination Process (SDP)-related shutdown events), there is insufficient experience to establish an accepted level of good practice. Therefore, for example, there is no accepted guidance based on experience of how many POSs are adequate to generate a reasonable estimate of LPSD risk. However, regarding NRC reviews of LPSD PRAs, design-specific LPSD PRAs have been reviewed by the staff for licensing new reactor design certifications and combined operating license applications as described in NUREG-0800, Chapter 19, Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors (NRC, 2012). For applications that require the average core damage frequency (CDF), it is necessary to define the average POSs. While it is possible at some level to characterize an average outage in terms of POSs, each outage will in fact be different in detail. The terminology "outage" is used synonymously with shutdown (SD) to represent the time which the reactor is shutdown and maintenance operations are usually being performed prior to the next operational cycle.

The sources of uncertainty identified were generally applicable to the use of a PRA to evaluate the average CDF and an outage specific risk assessment. However, for the latter, some, though not all, of the issues related to characterizing the POSs become moot since the specific plant configurations can be specified.

While the draft LPSD PRA standard (ANS, 2012) provides requirements for Qualitative Risk Assessment (QLRA) for SD operation using defense-in-depth principles, this was not addressed in the discussions. Similarly, the group did not address the risk from hazard groups other than internal events; while there may be some unique factors, the general perception was that the principal sources of model uncertainty not captured in the internal events section would be captured in the sections of the report dealing with the other hazard groups.

The sources of uncertainty identified are summarized in the Table C-1 in Appendix C categorized by the technical elements of a LPSD PRA as defined in the draft LPSD PRA standard (ANS, 2012). In Section 3.2,

, the number of issues identified during the session is presented. The table presents a list of LPSD PRA technical elements versus the type of issue raised (model uncertainty, completeness, level of detail, or parameter uncertainty) and shows the number of issues raised in a high, medium or low significance category which the session moderator, expert presenters, and participants assigned. In some cases, a clear identification as one type of uncertainty rather than another was not straightforward (see footnotes a, b, and c). In Sections 2.1.1, 2.1.2, 2.1.3, and 2.1.4, those issues that were classified as being of potentially high or medium significance are discussed in more detail for model uncertainty, completeness, level of detail, and parameter uncertainty respectively. The issues that were considered to be of low significance are not discussed further, but can be found in Appendix C, Table A-3.

5.2 Technical Elements of a LPSD PRA

Table 4 provides a summary of the number of issues identified during the workshop by PRA technical element (as defined in the draft LPSD PRA standard (ANS, 2012)) and by category

Table 4. LPSD PRA Technical Elements versus Uncertainty Category (Model, Completeness, Level of Detail, Parameter) and Workshop Technical Session Issue Importance Rank (High, Medium, Low)

Technical Element	Uncertainty Category			
	# of Model Uncertainties	# of Completeness Uncertainties	# of Level of Detail Uncertainties	# of Parameter Uncertainties
Plant operating state analysis (LPOS)		1 medium 1 medium-low	2 medium-low	
Initiating events analysis (LIE)	1 low [a]	1 low	1 medium	
Accident sequence development (LAS)	1 medium [a]		1 medium	
Success criteria (LSC)	1 medium	1 medium	1 low	
Systems Analysis (LSA)		1 low [b]		
Human Reliability analysis (LHR)	1 high 1 low		1 medium	
Data analysis (LDA)				2 low [c]
Quantification (LQU)	(No issues unique to LPSD PRA)			
LERF analysis (LLE)	1 medium	1 high 1 medium 1 low		

[a] Certain issues raised during the session for a particular technical element were difficult to categorize as a source of model uncertainty, completeness, level of detail, or parameter uncertainty. While these issues (Section 5.3.5 and 5.3.2, respectively) were classified as a source of model uncertainty, they would probably be manifested as a parameter uncertainty.

[b] This issue (Appendix C, Table A-3, Topic 13) raised here could have been classified as either a completeness or a level of detail issue.

[c] One of the issues could also have been classified as a level of detail issue (Appendix C, Table A-3, Topic 17), and the other as a model uncertainty issue (Appendix C, Table A-3, Topic 18).

5.3 LPSD PRA Model Uncertainties

The issues identified as sources of model uncertainty are summarized in this section, and the various identified sources of model uncertainty are presented in relation to the draft LPSD Standard technical element for which they are most closely relate. Identified sources of model uncertainty are discussed for five of the technical elements listed in Table 4.

5.3.1 HIGH – Human Reliability Analysis (HRA)

<u>Description of Issue</u>

HRA in general is recognized as a source of model uncertainty, but there are unique aspects of LPSD operations that create additional concerns as discussed here. There are significant differences between context for, and nature of, responses from those generally modeled for the at-power scenarios. Examples include:

- The guidance available to operators in the form of procedures for the low power POSs vs. the shutdown POSs in that there is no equivalent to the Emergency Operation Procedure (EOP) network for the latter; while there are abnormal procedures, they do not have the same characteristics. An EOP network refers to a method of explaining the way different sections of the EOPs interact and interconnect. Plants largely use abnormal operating procedures (AOPs) to respond to shutdown initiating events. AOPs generally do not receive the same level of verification and validation that the EOPs receive.
- For some plant disturbances leading to loss of a critical safety function, more problem solving and skill-of-the-craft or knowledge based response planning may be required than is typical for at-power situations.
- Errors of commission can be more significant for initiating events.
- Some of the scenarios may be very long-term scenarios, and thus repair and/or recovery of system functions can be more important.

Additionally, operator responses are relatively more important because many of the automatic means of responding to loss of a safety function are disabled.

Since the methods that have been developed for at-power HRA are largely focused on procedure driven responses with limited requirement for diagnosis, the applicability of these methods to the LPSD, but particularly the SD POSs needs to be examined further. Also, the

HRA methods generally do not address repair or recovery, since these are typically handled using actuarial data.

Specific issues that are identified as being unique to the modeling of LPSD include:

- treatment of dependency between at-initiator and post-initiator Human Failure Events (HFEs) (typically at power models don't address at-initiator HFEs with exception of those included in fault tree models for support system initiators)
- modeling of recovery and/or repair
- inclusion of specific errors of commission
- extendibility and applicability of at-power HRA models to SD conditions; is the performance shaping factor (PSF) coverage and the guidance for assessing the effect adequate

Additionally, there are issues that are relevant for at-power PRAs that may have an increased significance for shutdown conditions where there is increased reliance on manual actions. For example:

- Should there be a cutoff value for a single or for multiple Human Error Probabilities (HEPs) in an accident sequence cut set, and if so, what should it be?
- Should the cut-off value be variable depending on the context? This is particularly challenging if the time available for response is protracted.

<u>Manifestation in PRA</u>

HRA is critical to the evaluation of the LPSD PRA, and is reflected in the HFEs identified and defined and the HEPs evaluated for them.

<u>Relevance</u>

This is a significant issue for the base case PRA and for applications for existing and new reactor types if the design relies on manual actions to respond to shutdown events.

Because of the uncertain applicability of existing HRA methods for LPSD conditions, this model uncertainty has a moderate to high impact on the conclusions and risk insights and was categorized as HIGH significance. This source of model uncertainty is related to the Human Reliability Analysis technical element discussed in Topic 14, Appendix C, Table A-3.

Potential for R&D

There is a joint NRC/EPRI project underway to develop an improved approach to HRA. While the current phase of that project is focused on internal events, at-power PRAs, the work can be extended to address the LPSD specific context.

5.3.2 MEDIUM – Accident Sequence Development

Description of Issue

There is no consensus on how to model repair or recovery of a failed system. In at-power models, the consideration of repair or recovery of a failed system is rarely incorporated into the model, the typical exceptions being for offsite power and diesel generators. However, for the shutdown scenarios where the focus is on restoration of a critical safety function, there are typically fewer options for response, and the time scale of the accident sequences may be longer (see section 5.3.3 below), the modeling of repair or recovery may be more crucial. [Note that the term recovery is being used consistent with the definition in ASME/ANS RA-Sa-2009 (ASME/ANS, 2009), namely: "…restoration of a function lost as a result of a failed SSC by overcoming or compensating for its failure. Generally, modeled by using HRA techniques." In many cases, recovery options may not be proceduralized but may be more in the nature of workarounds that depend on the knowledge and skill of the operators. Such recovery actions create a challenge to existing HRA methods.

Manifestation in PRA

This is a form of model uncertainty since there is no consensus on what model to use to estimate the probability of repair or recovery, although the exponential model is commonly used, albeit without any technical basis. If the consensus of the community were to be that the exponential model was adequate, then this would be manifested by choosing a repair rate and characterizing the uncertainty in that rate, and this would become a parameter uncertainty.

Relevance

This is relevant to the base PRA and, therefore, potentially any application, and for existing and future reactors. Because this model uncertainty has a small to moderate impact on the conclusions and risk insights, it was categorized as MEDIUM significance. This source of model uncertainty is related to the Accident Sequence Development technical element discussed in Topic 9, Appendix C, Table A-3.

Potential for R&D

A database of repairs and recoveries made during shutdown could be useful.

5.3.3 MEDIUM – Success Criteria

Description of Issue

The selection of the mission time used in a PRA is a source of uncertainty. In constructing fault tree models for system unavailability, the assumption of 24 hours as a mission time is typically used as a default for at-power conditions. For both at-power and shutdown, some sequences may involve additional risk at later times. This can happen when there is reliance on one system to perform its function for a prolonged period of time when there are no clear alternative means of providing the same function. Specific mechanisms that may cause failures in addition to the usual mechanical failures that are typically taken into account include sump plugging//fuel assembly flow blockage concerns, for example. While this assumption is not unique to LPSD, it may have to be extended at least for some failure modes since failures after 24 hours may be significant. Random equipment failures after 24 hours are still not expected to be important.

Manifestation in PRA

This issue is manifested in the evaluation of system unavailability.

Relevance

This is of concern for both the base case PRA and for applications for existing and for new reactor types. Because this model uncertainty has a small to moderate impact on the conclusions and risk insights, it was categorized as MEDIUM significance. This source of model uncertainty is related to the Success Criteria technical element discussed in Topic 10, Appendix C, Table A-3.

Potential for R&D

This is classified as a model uncertainty because there are issues related to how to define a safe, stable state. When is it reasonable to assume that a failure of a critical safety function could be restored before core damage occurred? For example, is continuing on sump recirculation or feed and bleed for an extensive time realistic. Are there specific mechanisms for system failure that should be taken into account? How long can it be assumed that RWST can be refilled? This is related to Topic 1 in Table C-1 (see Appendix C). The model uncertainty, if resolved, would give an approach to resolving the level of detail issue addressed there.

5.3.4 MEDIUM – Large Early Release Frequency (LERF) Analysis

Description of Issue

Procedures such as accident management guidelines and security-related mitigation measures do not provide explicit instructions for response, but instead rely on decisions made by organizations such as the Technical Support Center who need to address known trade-offs between recovery event impacts ((e.g., recovery and restart of containment spray). This type of decision-making is not addressed well by current HRA methods which tend to be reflective more of the responses using EOPs. Additionally, Severe Accident Management Guidelines (SAMGs) and Extreme Damage Mitigation Guidelines (EDMGs) were developed with an event from at-power in mind. These guidance documents may not fit some shutdown POSs well.

Manifestation in PRA

PRA models may exclude such actions or assume they will be performed only when helpful. Inclusion of such actions when the actions mistakenly make things worse has not generally been included.

Relevance

This is a significant issue for the base case PRA and for applications for existing and new reactor types. Because this model uncertainty has a small to moderate impact on the conclusions and risk insights, it was categorized as MEDIUM significance. This source of model uncertainty is related to the Large Early Release Frequency Analysis technical element discussed in Topic 22, Appendix C, Table A-3.

Potential for R&D

This could be an extension of the HRA project discussed in 5.3.1. However, these types of decisions may not be easy to address probabilistically. Therefore, it could also be addressed by developing a philosophy on how best to deal with such circumstances when using a PRA in a decision-making process.

5.3.5 LOW – Initiating Events

Description of Issue

The availability and use of precursor events from plants other than that being analyzed is a potential source of uncertainty. The draft LPSD Standard (for capability category II) currently requires a review of plant specific events for the identification of potential initiating events, but much more useful information may be available from industry data. For example, an event which did not cause an initiator at the plant at which it occurred may have done so at the specific plant analyzed due to differences in the plant evolution, plant design, or plant operational practices. Use of such industry-wide data could be used to improve initiating event frequency data, by specializing the data to each plant and accounting for improvements in plant operations with time (e.g. adding additional level indication). However, this is contingent upon the availability of data and the level of detail that would allow such specialization. The specialization is likely to be a subjective process requiring assumptions to be made about the applicability of the data and its extrapolation.

Manifestation in the PRA

Failure to identify all potential initiating events leads to an incomplete model, but failure to account for the generic industry-wide experience would lead to an inaccurate assessment of initiating event frequencies. This issue could have been categorized an issue of completeness, as a form of model uncertainty related to the interpretation of data, or as a parameter uncertainty on initiating event frequencies.

Relevance

This issue is relevant for the analysis of existing reactors, but could also be relevant for advanced reactors, with the understanding that the interpretation of the impacts of the significance of a precursor would have to take into account the design differences in either of these cases. While it was concluded that this source of uncertainty would probably have a negligible to small impact on the conclusions and risk insights, and was therefore categorized as LOW significance, it is included here because the group considered that the expansion of a database to include such detail would benefit the future development of LPSD PRAs. This source of model uncertainty is related to the Initiating Events technical element discussed in Topic 6, Appendix C, Table A-3.

Potential for R&D

The comprehensiveness of LPSD PRAs would be enhanced by the compilation of a database with sufficient detail to allow the data to be reinterpreted for the target plant. In addition, completeness of modeling would be enhanced by providing guidance on extrapolating precursor data to plant-specific conditions.

5.4 LPSD PRA Completeness Uncertainty

In this section, the various identified sources of completeness uncertainty are presented in relation to the draft LPSD Standard technical element for which they are most closely related. Due to time constraints during the LPSD session, uncertainties with LOW significance were not discussed and thus not presented in this section. Identified sources of completeness uncertainty are discussed for three of the technical elements listed in Table 4 because only those three were identified as having MEDIUM or greater significance.

5.4.1 MEDIUM – Plant Operating State (POS)

Description of Issue

Some Level 1 scenarios end in a safe-stable state, such as successful feed and bleed, successful shutdown to terminate steam generator (SG) tube leak, or sump recirculation following a LOCA. These may lead to prolonged shutdown to allow for repair. While they are low frequency scenarios, the complete cycle to restoration of power is not generally modeled (see also Section 5.3.3).

Manifestation in PRA

These are essentially forced outages, so the concern is whether these are reflected in the assessment of the risk from forced outages.

Relevance

This is a medium significance issue for the base case PRA and for applications for existing and new reactor types. Because this completeness uncertainty has a small to moderate impact on the conclusions and risk insights, it was categorized as MEDIUM significance. This source of model uncertainty is related to the Plant Operational State Definitions technical element discussed in Topic 1, Appendix C, Table A-3.

Potential R&D

There is no specific need for R&D. However, guidance on determining representative forced outages and their schedules should address this issue in a coordinated manner. It is worth considering if these should be addressed in the context of forced outages.

5.4.2 MEDIUM – Success Criteria

Description of Issue

There is an insufficient research base of thermal-hydraulic (TH) analysis results for shutdown scenarios (SD) scenarios to give confidence that SD success criteria are accurately characterized. In addition, some have expressed concerns about the applicability of some codes. For example, can the codes analyze chugging effects? As a result, success criteria may be defined conservatively for selected conditions (e.g., no credit for SGs when reactor coolant system (RCS) is vented regardless of vent size).

Manifestation in PRA

Severe accident analyses for shutdown conditions impact accident sequence development, HEPs, success criteria, and the identification of severe accident event contributors. The concern is whether this is being done in an excessively conservative manner.

Relevance

This is an issue for the base case PRA and for applications for existing and new reactor types. Because this completeness uncertainty has a small to moderate impact on the conclusions and risk insights, it was categorized as MEDIUM significance. This source of model uncertainty is related to the Success Criteria technical element discussed in Topic 11, Appendix C, Table A-3.

Potential R&D

This could be classified as a completeness problem in that the knowledge base may not be large enough to cover all scenarios. There was some discussion that there is in principle no reason why some of the available codes cannot address the scenarios. Whenever a code is used, its limitations need to be recognized and reflected in the analysis. This may indeed lead to a conservative modeling in some cases. One aspect of the resolution would be a clear characterization of the applicability of the existing codes to shutdown conditions, and an assessment of their known limitations. Development of a more extensive database of case studies would also be valuable.

5.4.3 MEDIUM – Large Early Release Frequency (LERF) Analysis

Description of Issue

Three potential LERF contributors are excluded from the current list to be considered in Table 3.2.8-3 of the draft LPSD PRA standard (for PWRs) (ANS, 2012). The three omitted contributors are:

- hydrogen combustion (equipment survivability)
- steam explosions (reactor vessel (RV) head removed)
- induced residual heat removal (RHR) system failure (containment bypass)

Relating to the last bullet, the potential for RCS pressure boundary failure following a loss of all cooling may be of interest. In POSs when the RCS is still intact but is at lower initial pressure, the SGs may also be at atmospheric pressure and the lower decay heat present may mean that during subsequent RCS heatup, that a more uniform set of RCS temperatures occur after core uncovery that increases the potential for RCS pressure boundary failure relative to that seen for induced SG tube ruptures initiating from an accident initially at-power. RHR RV's failing open as the RCS pressurizes could lead to rapid overheating of the RHR system after core uncovery.

Manifestation in PRA

These contributors would affect the sequence development and contributors to LERF.

Relevance

This is an issue for the base case PRA and for applications for existing and new reactor types. Because this completeness uncertainty has a small to moderate impact on the conclusions and risk insights, it was categorized as MEDIUM significance. This source of model uncertainty is related to the Large Early Release Frequency Analysis technical element discussed in Topic 20, Appendix C, Table A-3.

Potential R&D

No solution was discussed.

5.5 LPSD PRA Level of Detail Issues

Several level of detail issues were identified in a number of the LPSD technical elements. However, since to some extent they are interrelated, unlike the model and completeness issues, they are discussed here as a group.

Description of Issue

There were a number of specific examples of issues related to level of detail, primarily associated with the number and characterization of POSs and grouping of initiating events. For example:

- For time-averaged models, quantifications are performed once for each POS. If the plant condition value changes within a POS, the time assumed for determining the decay heat/RCS level/RCS temperature and pressure within each POS can impact the computed response times and success criteria. This is possibly less important when considering CDF averaged over many evolutions rather than for a specific outage. This applies to the plant operating state analysis technical element. This source of model uncertainty is related to the Plant Operational State Definitions technical element discussed in Topic 2, Appendix C, Table A-3.
- Most LPSD models group forced outage evolutions by extent of RCS configuration changes required for repair rather than by a specific cause of outage. A representative cause of the outage type is then chosen. More severe causes, in terms of impact on mitigating systems, though low in frequency may be more risk significant. Typically the most frequent or common cause of each outage type is modeled as the cause of the outage ((e.g., refueling, loss of main feedwater, RCS seal LOCA, or SGTR). Exceedance of an AOT caused by a more severe impact on a mitigating system (loss of an emergency AC bus) is not chosen. This applies to the plant operating state analysis technical element. This source of model uncertainty is related to the Plant Operational State Definitions technical element discussed in Topic 4, Appendix C, Table A-3.
- Assumption of operating equipment failing at time of first demand eliminates development of sequences for conditions after start ((e.g., RHR relief valve is no longer isolated after pump start). Failure to credit RHR cooldown could lead to a similar omission. During SGTR, if RHR starts, subsequent RHR failure branches generally do not examine failures to isolate RHR allowing RCS depressurization through the RHR system following core uncover, i.e., potential bypass scenario. Same sequence applies to shutdown conditions when initially on RHR even though not following a SGTR. This applies to the accident sequence analysis technical element. This source of model uncertainty is related to the Accident Sequence Development technical element discussed in Topic 8, Appendix C, Table A-3.

- Incorporation of phenomenological conditions (e.g. RCS break location, "bounding" break sizes, access to high temperature locations at greater than boiling), and debris (NPSH, plugging) into the sequence models for each POS, particularly for temporary conditions resulting from testing or maintenance, can vary with specific maintenance activities and alignments, LOCA size and LOCA locations. LOCA locations are typically not distinguished as separate initiating events for PWRs during at-power but it may be more important to do so during shutdown. During shutdown a large frequency contributor to LOCAs are maintenance actions inadvertently diverting flow from the RCS. This applies to the accident sequence analysis or initiating events analysis technical element. This source of model uncertainty is related to the Success Criteria technical element discussed in Topic 12, Appendix C, Table A-3.
- There are specific configuration, spatial, or environmental conditions that can affect system availability or long-term reliability that may be different in different POSs. Examples include: temporary removal of flood barriers or fire barriers; reconfiguration of ventilation; instrument tube bolt detensioning with RCS not yet vented; NPSH concerns; plugging from debris in the shutdown following a LOCA); specific unusual system alignments. Identification of these conditions is more difficult than for at-power because of the many POSs and parallel activities going on. Such conditions may affect the feasibility of systems performing their function once an accident begins. Of particular concern are system conditions at the time of an RCS repressurization accident. This applies to the systems analysis technical element. This source of model uncertainty is related to the Systems Analysis technical element discussed in Topic 13, Appendix C, Table A-3.
- Concern about the completeness in identifying at-initiator HFEs via reviews of industry operating experience and related reviews of plant specific test and maintenance activities as part of the pre-initiator HFE evaluation process. The number of procedures available for review is huge and the search criteria for identifying such HFEs are not well established. Historical records are substantial but not always sufficiently documented to extrapolate their applicability to other plants. Further, use of a pre-defined set of initiators dissuades analysts from examining individual causes within those defined and thereby account for plant specific unique boundary categories conditions. This applies to the human reliability analysis technical element. This source of model uncertainty is

related to the Human Reliability Analysis technical element discussed in Topic 16, Appendix C, Table A-3.

Manifestation in PRA

Keeping the level of detail high will generally lead to conservatism since limiting cases will be adopted as the reference cases for a POS or an initiating event. However, there are circumstances where increasing the level of detail could identify new, challenging scenarios that could lead to an increase in the calculated risk.

Relevance

This is applicable to PRAs of existing and future plants and affects both the base case and applications.

Potential R&D

As more experience is gained with developing and using LPSD PRAs, it might be appropriate to document good practices such that the appropriate level of detail will become more widely appreciated.

5.6 LPSD PRA Parameter Uncertainties

No medium or high parameter uncertainty issues were identified. Methods for estimating parameters and characterizing their uncertainty are well established, and while the data may be more sparse for some aspects of LPSD operations, the issues related to their estimation are not unique to LPSD PRA.

6. UNCERTAINTIES IDENTIFIED FOR LEVEL 2 AT-POWER PRA

6.1 Introduction

There were many topics raised for discussion during the session on Level 2 At-Power PRAs. One significant conclusion was that the extension of PRA models beyond a Large Early Release Frequency (LERF) model to a full Level 2 PRA model with multiple release characteristics introduces many more potential sources of uncertainty than a LERF-only model. That is, there is a large amount of literature and in many cases significant consensus in the industry for issues associated with LERF, but there are a significant number of nuances and new issues that arise when the scope is expanded to a full Level 2 PRA model.

It should be noted that the scope of the discussions was limited to at-power conditions and an attempt was made to account for new reactor designs when feasible in the characterization of the sources of uncertainty. Additionally, the characterization of the importance also considered accident strategy management development as many of the issues raised could influence the acceptability of identified long-term mitigation strategies.

The sources of uncertainty identified are summarized in Appendix D, Table A-4. In this table, the sources of uncertainty are categorized by the technical elements of the Level 2 PRA as defined in the draft Level 2 PRA standard. In Section 6.2, the number of issues identified is presented as a function of technical element and by type. In some cases, a clear identification as one type of uncertainty rather than another was not straightforward. In Sections 6.3, 6.4, 6.5, and 6.6 those issues that were classified as being of potentially HIGH or MEDIUM significance are discussed in more detail for model uncertainty, completeness, level of detail, and parameter uncertainty, respectively. Ultimately, however, most of the issues identified in the Level 2 session were characterized as sources of model uncertainty with a few of the issues captured as level of detail or completeness issues. There were no parameter uncertainty issues specifically identified as such. The sources of uncertainty that were characterized as being of low significance are not discussed in detail, but are included in Appendix D, Table A-4 for completeness.

6.2 Technical Elements of a Level 2 At-Power PRA

Table 6.1 provides a summary of the number of issues identified during the workshop by PRA technical element (as defined in the draft Level 2 PRA standard (ANSI/ANS/ASME, 2010)) and by category.

Table 5. Level 2 PRA Technical Elements versus Uncertainty Category (Model, Completeness, Level of Detail, Parameter) and Workshop Technical Session Issue Importance Rank (High, Medium, Low)

Technical Element	Uncertainty Category			
	# of Model Uncertainties	# of Completeness Uncertainties	# of Level of Detail Issues	# of Parameter Uncertainties
Level 1 / Level 2 Interface (L1)		1 high	1 high 1 low	
Containment Capacity Analysis (CP)	1 high 4 medium 2 low			
Severe Accident Progression Analysis (SA)	6 high 4 medium 2 low			
Probabilistic Treatment (PT)	4 high 2 medium			
Source Term Analysis (ST)	2 medium			

6.3 Level 2 At-Power PRA Model Uncertainties

The issues identified as sources of model uncertainty are summarized in this section. As was previously noted, the large majority of the identified issues during the workshop were characterized as model uncertainty issues rather than completeness, level of detail, or parameter uncertainty issues. The potential high sources of model uncertainty are presented first in subsection 6.3.1 through 6.3.11. The identified medium sources of model uncertainty are presented in subsections 6.3.12 through 6.3.24. The low sources of model uncertainty are only listed in Appendix D, Table A-4 and are not discussed in more detail here.

6.3.1 HIGH – Dynamic Load Impacts on Containment Failure Mode

Description of Issue

Severe accidents can lead to environment conditions beyond those considered during the design of the containment system.

Containment failure mechanisms caused by (or influenced by) accident phenomena/conditions such as the following should be considered:

- hydrogen combustion (deflagration and detonation)
- hydrodynamic loads
- interactions between molten core debris and water

The containment response is highly dependent on the geometry and definition of the impulse.

Manifestation in the PRA

These issues are typically handled with separate engineering analysis. In some cases, a bounding treatment may be sufficient to show that the probability of failure is low. If this is not the case, then the potential to become an important source of model uncertainty increases.

Relevance

This is a potential high source of model uncertainty for the analysis of existing reactors, and could also be relevant for advanced reactors. The issue is applicable to the base model and could also be important in certain applications of the model. Because this model uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as having a HIGH significance. This source of model uncertainty is related to the Containment Capacity Analysis technical element discussed in Topic 7, Appendix D, Table A-4.

Potential for R&D

Research and development could aid in developing a middle ground between simplified hand calculations and complex methods such as computational fluid dynamic solutions for shock wave propagation or nonlinear dynamic explicit finite element analysis for estimating the structural effects, which could be useful to provide realistic estimates of the dynamic load impacts on containment.

6.3.2 HIGH - Thermally Induced Failure of RCS Pressure Boundary

Description of Issue

For PWRs, this issue is associated with thermally induced failures under high pressure conditions of hot leg piping/vessel nozzles, surge lines or steam generator tubes.

The probability of a thermally induced rupture of steam generator tubes depends on several factors including:

- treatment of natural circulation and loop seal clearing
- thermal-hydraulic conditions (temperature and pressure) in the RCS and steam generators,
- material properties impacting creep rupture
- presence of defects in the steam generator tubes

For BWRs, the severe accident progression can also result in thermally induced failures of the pressure boundary. Failure of the RCS can lead to RPV depressurization prior to vessel breach, and depending on the failure location, can have a significant impact on fission product transport and release. Key issues to consider include:

- treatment of the SRV stochastic failure probability due to cycling demands at high RPV pressure (a stuck open SRV would depressurize the RPV and lead to fission product transport to the suppression pool)
- material properties impacting creep rupture of the main steam line (main steam line failure would lead to bypass of the suppression pool)

Manifestation in the PRA

A large amount of information is available to support failure likelihoods for existing PWR fleet. However, it may be a potentially larger source of uncertainty for unanalyzed reactor designs.

For BWRs, the timing of a stuck open relief valve versus continued heatup of the main steam line leading to failure can have a significant impact on fission product transport and release.

Relevance

This is a potential high source of model uncertainty for the analysis of existing reactors, and could also be relevant for advanced reactors. The issue is applicable to the base model and could also be important in certain applications of the model. Because this model uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as having a HIGH significance. This source of model uncertainty is related to the Severe Accident Progression Analysis technical element discussed in Topic 11, Appendix D, Table A-4.

Potential for R&D

Research and development could aid in better addressing this issue especially in the area of new reactors.

6.3.3 HIGH - Recovery of a Degraded Core

Description of Issue

Phenomenological issues associated with in-vessel core melt progression and retention are highly complex and uncertain. Important issues include:

- cladding oxidation behavior

- fuel and clad melting and relocation mechanisms
- crust formation/crust failure in the lower portions of the fuel

Manifestation in the PRA

These issues impact the potential for recovery of a degraded core. In many Level 2 PRAs, credit for in-vessel accident mitigation has been modeled for sequences where water flow was restored within a short period of time of the onset of core damage and prior to significant core geometry changes.

Relevance

This is a potential high source of model uncertainty for the analysis of existing reactors, and could also be relevant for advanced reactors. The issue is applicable to the base model and could also be important in certain applications of the model. The impacts from recovery of a damaged core could impact the development of appropriate accident management strategies. Because this model uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as having a HIGH significance. This source of model uncertainty is related to the Severe Accident Progression Analysis technical element discussed in Topic 13, Appendix D, Table A-4.

Potential for R&D

Focused sensitivity studies and additional research might be warranted.

6.3.4 HIGH - External Cooling of RPV Lower Head

Description of Issue

The conditions associated with a molten pool in the lower head region are very uncertain. The ability to model side failure, unzipping, localized attack, or penetration failure depend on the nature of the pool or debris. For some plants, there is uncertainty as to whether the vessel can be cooled externally.

The issues to assess include:

- Whether the imposed heat flux exceeds the heat removal capability (critical heat flux) on the external surface.
- The potential for melting of the vessel wall under the thermal loading from the molten pool.

- The pressure bearing capability of the vessel wall held at high temperature inside, and low temperature outside.

<u>Manifestation in the PRA</u>

Depending on the reactor design, ex-vessel cooling of the RPV lower head may or may not be credited in the Level 2 analysis.

<u>Relevance</u>

This is a potential high source of model uncertainty for those designs that credit this means of averting vessel failure as it would have a significant impact on accident sequence progression and development of accident management strategies. Because this model uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as having a HIGH significance. This source of model uncertainty is related to the Severe Accident Progression Analysis technical element discussed in Topic 15, Appendix D, Table A-4.

<u>Potential for R&D</u>

Focused sensitivity studies and additional research might be warranted.

6.3.5 HIGH - Ex-Vessel Fuel-Coolant Interactions

<u>Description of Issue</u>

The calculated loads for ex-vessel FCI events may exceed the structural capacity of a typical cavity (i.e., sub-cooled water, lower pressure, weaker structure than the vessel).

Ex-vessel FCIs may also impact accident progression and fission product release by:

- debris transport outside of cavity and/or pedestal
- enhanced hydrogen production
- releases of radioactive material

<u>Manifestation in the PRA</u>

The potential effect of structural failures in cavity/pedestal region on containment integrity may be unclear for some reactor designs. However, there could be a longer term beneficial effect related to debris coolability and reduced core-concrete attack.

Relevance

Plant-specific susceptibility to this issue could affect the SAMG strategy. Because this model uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as having a HIGH significance. This source of model uncertainty is related to the Severe Accident Progression Analysis technical element discussed in Topic 17, Appendix D, Table A-4.

Potential for R&D

Focused sensitivity studies and additional research might be warranted.

6.3.6 HIGH - Energetic Burning of Hydrogen and Combustible Gases

Description of Issue

The quasi-static and dynamic loads imposed on the containment structure as a result of hydrogen and combustible gas burns is impacted by a number of factors including:

- mixing and/or stratification of the containment atmosphere
- extent of steam inerting
- propagation of ignition and deflagration flames
- flame acceleration and transition from deflagration to detonation
- ignition sources
- heat losses to structures

Manifestation in the PRA

There is a relatively good understanding of flammability limits and the thresholds for deflagration and detonation for hydrogen. There is less of an understanding for the combination of hydrogen and carbon monoxide. There is also limited understanding of conditions resulting in transition to detonation for these conditions. Additionally, the potential effect of structural failures outside of the containment has not been thoroughly studied (e.g., reactor buildings). All of these issues impact assumptions in the Level 2 PRA model development regarding accident sequence progression.

Relevance

This is a potential high source of model uncertainty as it could have a significant impact on accident sequence progression and development of accident management strategies. Because this model uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as having a HIGH significance. This source of model uncertainty is related to the Severe Accident Progression Analysis technical element discussed in Topic 20, Appendix D, Table A-4.

Potential for R&D

Research and development could aid in developing specialized tools for determining hydrogen and combustible gas distribution and examining the impacts of flame and wave propagation.

6.3.7 HIGH - Impact of Core Debris / Concrete Interactions

Description of Issue

Core debris concrete attack can result in:

- undermining of containment structures (cavity walls/vessel support) by the core debris
- generation of non-condensable gas ($H_2/CO/CO_2$)
- lateral spreading of debris and potential for contact with containment pressure boundary
- potential for groundwater and environmental releases of radioactive material

Manifestation in the PRA

The details of core debris concrete attack discussed above all have an impact on the assumptions utilized in the Level 2 PRA model development regarding accident sequence progression.

Relevance

This was identified as a potential high source of model uncertainty. There is a potential large impact on the magnitude and type of late releases and land contamination issues. Because this model uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as having a HIGH significance. This source of model uncertainty is related to the Severe Accident Progression Analysis technical element discussed in Topic 22, Appendix D, Table A-4.

Potential for R&D

No specific areas for potential R&D were identified. However, potential research could focus on determining if these uncertainties in core debris concrete attack actually do significantly affect the large release frequency.

6.3.8 HIGH - Modeling of Operator Actions During Severe Accidents

Description of Issue

There is no consistent approach for treating human error probabilities using a methodology that is consistent with the framework of the Level 1 PRA analysis (i.e., nominal values with performance shaping factors). In addition, there are unique influences that should be considered in the development of the Level 2 operator actions, such as:

- reluctance to make any decision that directly results in a release
- communication and decision-making between the control room, technical support center (TSC), and emergency operations facility (EOF)
- parsing of failed versus reliable indication

Manifestation in the PRA

The human error probability events are an integral part of any PRA model development process.

Relevance

This was identified as a potential high source of model uncertainty. It is recognized as a generic source of model uncertainty that is applicable to the base model and applications for existing reactors and new reactors. Because this model uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as having a HIGH significance. This source of model uncertainty is related to the Probabilistic Treatment technical element discussed in Topic 24, Appendix D, Table A-4.

Potential for R&D

Focused sensitivity studies and additional research might be warranted.

6.3.9 HIGH - Treatment of SAMG (and Other Accident Management) Actions

Description of Issue

This affects HRA and accident progression analysis portions of the Level 2 PRA. Identified issues include:

- Most SAMG actions inherently have a positive and negative effect (e.g., containment sprays reduce containment pressure but increases likelihood of a hydrogen deflagration).
- Focusing only on "important" post-core damage operator actions may not be sufficient if the goal is to be best-estimate.

Manifestation in the PRA

The human error probability events derived from the SAMGs are an integral part of any PRA model development process. Additionally, the impacts of these actions could influence the severe accident progression analysis.

Relevance

This was identified as a potential high source of model uncertainty. Similar to the HRA issue, it is applicable to the base model and applications for existing reactors and new reactors. Because this model uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as having a HIGH significance. This source of model uncertainty is related to the Probabilistic Treatment technical element discussed in Topic 25, Appendix D, Table A-4.

Potential for R&D

There is a joint NRC/EPRI project underway to develop an improved approach to HRA. While the current phase of that project is focused on internal events, at-power PRAs, and EOP/AOP type procedures, it is anticipated that an extension to consider SAMGs will be pursued at some point in the future.

6.3.10 HIGH - Equipment / Instrument Survivability for SAMG Implementation

Description of Issue

This issue affects both explicit (e.g., equipment availabilities) and implicit (e.g., assumptions about available indication and its effects on operator response) pieces of the Level 2 PRA. The

Level 2 PRA model assessment would need to consider not only pressure, temperature, humidity, and radiation impacts, but also the potential effects of hydrogen transport and deflagration/detonation into the reactor building or auxiliary building.

Manifestation in the PRA

Current approaches typically provide limited credit for equipment and instrumentation in severe environments. Some credit for systems under severely degraded conditions could improve risk profile and realism.

Relevance

This was identified as a potential high source of model uncertainty. It is applicable to the base model and applications for existing reactors and new reactors. Because this model uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as having a HIGH significance. This source of model uncertainty is related to the Probabilistic Treatment technical element discussed in Topic 26, Appendix D, Table A-4.

Potential for R&D

No specific areas for potential R&D were identified.

6.3.11 HIGH - Passive System Reliability

Description of Issue

Some of the new reactor designs are relying on passive features to mitigate and/or reduce the impacts of a potential severe reactor accident. Definitive knowledge about the reliability of these systems and how that is factored into the Level 2 PRA model development process may not be well established.

Manifestation in the PRA

This is a potential high source of model uncertainty for those designs that utilize passive systems as a means of averting vessel or containment failure as it would have a significant impact on accident sequence progression and development of accident management strategies.

Relevance

This was identified as a potential high source of model uncertainty especially for some new reactor designs. Because this model uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as having a HIGH significance. This source of model uncertainty is related to the Probabilistic Treatment technical element discussed in Topic 29, Appendix D, Table A-4.

Potential for R&D

There is potential to reduce the associated uncertainty with focused research and experiments.

6.3.12 MEDIUM - Containment Failure Modes Given Quasi-Static Loads

Description of Issue

The mode and location of containment leakage and failure is one of the most important parameters impacting the magnitude and timing of radionuclide release. Multiple approaches have been taken to assessing containment failure mode and location including use of failure information from similar plants and detailed structural analyses. The analysis also needs to account for material creep/degradation due to high temperatures.

Manifestation in the PRA

Different modes of containment failure can typically be factored into the Containment Event Tree (CET) structure.

Relevance

This was identified as a medium source of model uncertainty. It is applicable to the base model and applications for existing reactors and new reactors. Because this model uncertainty has a small to moderate impact on the conclusions and risk insights, it was categorized as MEDIUM significance. This source of model uncertainty is related to the Containment Capacity Analysis technical element discussed in Topic 6, Appendix D, Table A-4.

Potential for R&D

No specific areas for potential R&D were identified.

6.3.13 MEDIUM - Indirect Mechanisms of Containment Failure

Description of Issue

Severe accident phenomenon may lead to containment integrity challenges in addition to high static or dynamic pressures. Mechanisms such as those discussed below may also challenge containment integrity:

- Debris concrete interactions have the potential to result in reactor cavity/pedestal structural failure.
- RPV lower head failure under high pressure conditions may result in reactor cavity/pedestal structural failure.
- Ex-vessel steam explosion may potentially cause reactor cavity/pedestal structural failure.
- Seismic induced leakage may occur (e.g., through penetrations) for well-beyond design basis earthquakes

<u>Manifestation in the PRA</u>

Different mechanisms for containment failure can typically be factored into the CET structure.

<u>Relevance</u>

It is applicable to the base model and applications for existing reactors and new reactors. Depending on the basis that is established for each of these issues and the associated importance measures for specific applications, sensitivity cases may be warranted to examine the potential impacts from alternate assumptions (i.e., different failure likelihoods) associated with each of these issues. Because this model uncertainty has a small to moderate impact on the conclusions and risk insights, it was categorized as MEDIUM significance. This source of model uncertainty is related to the Containment Capacity Analysis technical element discussed in Topic 8, Appendix D, Table A-4.

<u>Potential for R&D</u>

No specific areas for potential R&D were identified.

6.3.14 MEDIUM - Quasi-Steady Failure Threshold Methods and Correlation Between Failure Pressure and Leak Rate

<u>Description of Issue</u>

The ability to determine failure pressure given defined conditions has significant uncertainty, especially for concrete containments. Significant uncertainties are also associated with construction detail and ageing effects. The basis for developing fragility curves is typically subjective.

Manifestation in the PRA

Ultimately, the containment failure capacity is typically characterized by a point estimate (e.g., lower bound or "best" estimate pressure) or by a probability density function (fragility curve).

Relevance

It is applicable to the base model and applications for existing reactors and new reactors. Because this model uncertainty has a small to moderate impact on the conclusions and risk insights, it was categorized as MEDIUM significance. This source of model uncertainty is related to the Containment Capacity Analysis technical element discussed in Topic 9, Appendix D, Table A-4.

Potential for R&D

There is a continued need for development in this area. Note that recent and ongoing effort in this area has arisen from NRC-sponsored work at Sandia National Labs, as well as ongoing collaboration between the NRC and the Atomic Energy Regulatory Board of India. A key issue is the state-of-practice in translating finite element model results (liner stresses, strains, and deformations) in to functions describing containment leakage area (or rate) as a function of pressure. Note that this item relates to the following issue discussed in Section 6.3.15.

6.3.15 MEDIUM - Containment Failure Characteristics

Description of Issue

Given containment failure occurs, the source of uncertainty relates to how the "final" containment failure is characterized (i.e., location and size). The containment failure size could be a function of containment load (e.g. pressure) or timing (e.g., time when debris contacts the liner). The containment failure could also be characterized by a threshold model or a leak before break model.

The threshold model defines a threshold pressure at which the containment is expected to fail with a large breach. In the leak before break model, containment leakage is expected to precede a major rupture and the leakage rate is modeled to increase with increasing internal containment pressure up to the ultimate capability pressure, at which point a larger failure of the containment is expected to occur.

If the rate of addition of mass and energy to the containment atmosphere is smaller than or equal to the leakage rate, further containment pressurization is not expected to occur and catastrophic failure of the containment may be averted.

Manifestation in the PRA

Different characteristics of containment failure can be factored into the CET structure.

Relevance

It is applicable to the base model and applications for existing reactors and new reactors. Because this model uncertainty has a small to moderate impact on the conclusions and risk insights, it was categorized as MEDIUM significance. This source of model uncertainty is related to the Containment Capacity Analysis technical element discussed in Topic 10, Appendix D, Table A-4.

Potential for R&D

No specific areas for potential R&D were identified.

6.3.16 MEDIUM - RPV Lower Head Failure Mechanism

Description of Issue

Alternative lower head failure mechanisms should be considered such as:

- global (creep) failure of reactor pressure vessel
- local failure of lower head of reactor pressure vessel (e.g. at lower head penetrations)
- early RPV leakage via failed open instrument tubes (also leading to a potential bypass of containment for some designs)

Manifestation in the PRA

Different RPV failure mechanisms can be factored into the CET structure.

Relevance

It is applicable to the base model and applications for existing reactors and new reactors. Because this model uncertainty has a small to moderate impact on the conclusions and risk insights, it was categorized as MEDIUM significance. This source of model uncertainty is related to the Severe Accident Progression Analysis technical element discussed in Topic 14, Appendix D, Table A-4.

Potential for R&D

No specific areas for potential R&D were identified.

6.3.17 MEDIUM - In-Vessel Hydrogen Generation

Description of Issue

The extent of in-vessel hydrogen generation is believed to be sensitive to a number of parameters including:

- the extent of in-core flow blockage
- clad ballooning
- recovery and addition of water
- relocation of molten fuel
- zirconium melt breakout temperature
- fuel rod collapse temperature
- fractional local dissolution of UO_2 in molten Zirconium
- melt relocation heat transfer coefficient
- particulate debris characteristic size following core collapse
- particulate debris characteristic size following relocation to lower plenum
- porosity of fuel debris beds

Manifestation in the PRA

These issues are typically addressed via the code used for accident sequence progression analysis (e.g., MAAP or MELCOR). However, the uncertainty arises in the actual amount of total hydrogen that is generated for each sequence type and how that is factored into the Level 2 PRA model.

Relevance

It is applicable to the base model and applications for existing reactors and new reactors. Because this model uncertainty has a small to moderate impact on the conclusions and risk insights, it was categorized as MEDIUM significance. This source of model uncertainty is related to the Severe Accident Progression Analysis technical element discussed in Topic 19, Appendix D, Table A-4.

Potential for R&D

No specific areas for potential R&D were identified.

6.3.18 MEDIUM - Ex-Vessel Debris Bed Coolability

Description of Issue

The coolability of core debris relocated to the reactor cavity/pedestal regions is subject to a number of uncertainties. One of the most important is the effective upward heat flux to an overlying water pool.

A critical question is whether water penetration through the upper debris bed surface (e.g., through cracks) will facilitate heat transfer at rates above conduction limited heat transfer through a solid crust.

There are also still substantial uncertainties in the two-dimensional cavity erosion profiles (i.e., heat flux partitioning between axial and radial directions). Note that excessive radial erosion can undermine containment integrity, while excessive axial erosion can fail the basemat, leading to ground contamination and release of radiological source terms in the environment.

Manifestation in the PRA

If the debris bed is not coolable, then there is potential for a large impact on magnitude and type of late releases and land contamination issues.

Relevance

It is applicable to the base model and applications for existing reactors and new reactors. Because this model uncertainty has a small to moderate impact on the conclusions and risk insights, it was categorized as MEDIUM significance. This source of model uncertainty is related to the Severe Accident Progression Analysis technical element discussed in Topic 21, Appendix D, Table A-4.

Potential for R&D

No specific areas for potential R&D were identified.

6.3.19 MEDIUM - Ex-Vessel Hydrogen and Combustible Gas Generation

Description of Issue

The extent of ex-vessel hydrogen and combustible gas production during core concrete interactions is impacted by a number of uncertain parameters including:

- ex-vessel debris/water heat transfer parameters
- enhancements to upward heat transfer by penetration of overlying water into cracks and fissures in the debris crust
- extent of sideways versus downwards concrete erosion
- concrete aggregate material composition
- quantity of remaining metals in the melt (zirconium and steel)

Manifestation in the PRA

These issues are typically addressed via the code used for accident sequence progression analysis (e.g., MAAP or MELCOR). However, the uncertainty arises in the actual amount of total combustible gas generation that occurs for each sequence type and how that is factored into the Level 2 PRA model.

Relevance

This was identified as a medium source of model uncertainty. It is applicable to the base model and applications for existing reactors and new reactors. Because this model uncertainty has a small to moderate impact on the conclusions and risk insights, it was categorized as MEDIUM significance. This source of model uncertainty is related to the Severe Accident Progression Analysis technical element discussed in Topic 23, Appendix D, Table A-4.

Potential for R&D

No specific areas for potential R&D were identified.

6.3.20 MEDIUM - Random and/or Seismically Induced Failure Probabilities Not Covered in Level 1 PRA Data Collection

Description of Issue

Unlike Level 1 PRA, equipment used in the severe accident management guidelines (SAMGs) often does not have the necessary data to support data-informed failure probability assignment.

Manifestation in the PRA

The failure probabilities are represented as inputs to the basic events used to represent the failure modes of the Level 2 PRA model credited systems.

Relevance

It is applicable to the base model and applications for existing reactors and new reactors. Because this model uncertainty has a small to moderate impact on the conclusions and risk insights, it was categorized as MEDIUM significance. This source of model uncertainty is related to the Probabilistic Treatment technical element discussed in Topic 27, Appendix D, Table A-4.

Potential for R&D

No specific areas for potential R&D were identified. This issue should improve over time as equipment credited in the Level 2 analysis becomes more mainstream.

6.3.21 MEDIUM - Correlation Introduced by Common Physical Parameters

Description of Issue

NUREG-1855 discusses one type of correlation, the state-of-knowledge correlation (SOKC) or epistemic correlation which arises when the same parameter uncertainty model is used to quantify the probabilities of two or more basic events.

Another type of correlation relates to phenomenological events which are correlated through dependencies on other common causal events/parameters. For example, in-vessel radionuclide release and hydrogen generation are not independent but correlated through dependencies on common accident progression parameters.

Manifestation in the PRA

This issue should not be important for models which are relying on the use of point estimate mean values, but a method for including these dependencies appropriately in a parametric uncertainty analysis has not been defined.

Relevance

It is applicable to the base model and applications for existing reactors and new reactors. Because this model uncertainty has a small to moderate impact on the conclusions and risk insights, it was categorized as MEDIUM significance. This source of model uncertainty is related to the Probabilistic Treatment technical element discussed in Topic 28, Appendix D, Table A-4.

Potential for R&D

Research and development on developing a method to account for these types of dependencies in the parametric uncertainty analysis might be warranted.

6.3.22 MEDIUM - Source Term Characteristics

Description of Issue

In addition to uncertainties introduced by uncertainties in the accident progression phenomena additional uncertainties exist for radionuclide formation, transport and deposition related to:

- in-vessel fission product release
- fission product retention in the RCS and containment
- fission product chemistry
- fission product release during core debris concrete interactions
- late revolatization from the RCS and containment
- fission product scrubbing in water pools
- fission product revolatization from water pools
- fission product grouping

Manifestation in the PRA

The understanding of fission product behavior is improving but there are still significant uncertainties. This is more of a long-term health effect issue for Level 3 analysis.

Relevance

It is applicable to the base model and applications for existing reactors and new reactors. Because this model uncertainty has a small to moderate impact on the conclusions and risk insights, it was categorized as MEDIUM significance. This source of model uncertainty is related to the Source Term Analysis technical element discussed in Topic 30, Appendix D, Table A-4.

Potential for R&D

No specific areas for potential R&D were identified.

6.3.23 MEDIUM - Source Term Attenuation in Structures Outside the Primary Containment

Description of Issue

Secondary containment/auxiliary building may represent an additional effective retention area for radionuclides for certain types of sequences or containment leakage failure modes. For example, prior PRAs have credited auxiliary/safeguards buildings for fission product attenuation for ISLOCA containment bypass sequences.

Uncertainties arise as a result of the structural capacities of these structures (many have blowout panels, low pressure ducting, etc.), the impacts of potential phenomenological events in these structures (e.g. hydrogen burns) and the assessment of the release pathways from these structures.

Manifestation in the PRA

Some impacts on short-term releases, but more important for long term health effects.

Relevance

It is applicable to the base model and applications for existing reactors and new reactors. Because this model uncertainty has a small to moderate impact on the conclusions and risk insights, it was categorized as MEDIUM significance. This source of model uncertainty is related to the Source Term Analysis technical element discussed in Topic 31, Appendix D, Table A-4.

Potential for R&D

No specific areas for potential R&D were identified.

6.3.24 MEDIUM - Impact from Accident Duration Truncation of Sequence Runs

Description of Issue

Truncating deterministic accident progression simulations at (e.g., 48 hours) could non-conservatively bias results toward risk from earlier releases.

Assumption that recovery actions are 100% effective after some time may not provide the best estimate presentation of the results.

Manifestation in the PRA

There are some impacts on short-term releases, but this issue is more important for long-term health effects.

Relevance

It is applicable to the base model and applications for existing reactors and new reactors. Because this model uncertainty has a small to moderate impact on the conclusions and risk insights, it was categorized as MEDIUM significance. This source of model uncertainty is related to the Source Term Analysis technical element discussed in Topic 32, Appendix D, Table A-4.

Potential for R&D

No specific areas for potential R&D were identified.

6.4 Level 2 At-Power PRA Completeness Uncertainty

The issues identified as completeness issues are summarized in this section. The only identified completeness issue stemmed from the Level 1 / Level 2 Interface (L1) technical element.

6.4.1 HIGH - Partial Degraded Performance Not Credited in Level 1

Description of Issue

Numerous Level 1 PRA modeling choices can be influenced by the go / no-go nature of Level 1 PRA end-states. In some cases, partial flow from systems or injection flow from lower capacity systems not credited in the Level 1 PRA model can have an adverse impact on the severe accident progression. Note that this issue could also apply to the data collection efforts where a degraded flow condition is counted as a failure, but in fact could be equivalent to some lower capacity systems.

For instance, injection of water into a degraded core might be able to prevent vessel failure, but there is also the potential for increased fuel–coolant interactions leading to additional releases of hydrogen and fission products.

Manifestation in the PRA

Failure to identify all potential impacts from partial or degraded flow scenarios could lead to an inaccurate assessment of source term characteristics. This also could have been categorized as an as a form of model uncertainty related to the interpretation of data, or as a parameter uncertainty on failure probabilities, but was assigned to the completeness category since most Level 2 PRA models do not consider the potential negative impacts of partial flow conditions.

Relevance

This is a potential high source of uncertainty relevant for the base case analysis of existing reactors, and could also be relevant for advanced reactors. The issue could also be relevant in certain applications of the Level 2 PRA model when differences in non-LERF source term characteristics drive the results. Because this completeness uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as having a HIGH significance. This source of model uncertainty is related to the Level 1 / Level 2 Interface technical element discussed in Topic 3, Appendix D, Table A-4.

Potential for R&D

The comprehensiveness of Level 2 At-Power PRAs would be enhanced by focused thermal-hydraulic analysis to determine if indeed partial or degraded flow could significantly alter the source term characteristics.

6.5 Level 2 At-Power PRA Level of Detail Uncertainty

The issues identified as level of detail issues are summarized in this section. The only identified level of detail issues stemmed from the Level 1 / Level 2 Interface (L1) technical element. One of the issues was characterized as a low source of uncertainty and one of the issues was identified as a potential high source of uncertainty. Only the potential high source of uncertainty is discussed here. The low sources of uncertainty are only listed in Appendix D, Table D-1.

6.5.1 HIGH - Number of Plant Damage State Groups

Description of Issue

Grouping accident sequences or cutsets from the Level 1 PRA into plant damage states for input into the Level 2 PRA potentially introduces uncertainties due to the resulting loss of modeling detail.

Care should be taken to ensure that information is not lost due to PDS simplifications. The following questions should be considered:

- Is the number of PDS groups sufficient to represent the significant differences among the Level 1 sequences?
- If fewer PDS groups are used, does the "representative" sequence reasonably bound the set of sequences assigned to the PDS?
- Are the intergroup sequence characteristics sufficiently similar such that the representative sequence does not create an overly conservative or non-conservative bias in the modeling?

Manifestation in PRA

Proper treatment of these issues is covered by the Level 2 PRA Standard. Therefore, appropriate model development should sufficiently address this issue. A higher level of detail will generally lead to conservatism since limiting cases will be adopted for the reference cases for a PDS. However, there could be some applications of the model where expansion of the plant damage states or representative scenarios may be warranted so that the PDS grouping does not adversely skew the results.

Relevance

This is applicable to PRAs of existing and future plants and could be more of an issue in applications than in the base PRA model. Because this level of detail uncertainty has a moderate to high impact on the conclusions and risk insights, it was categorized as having a HIGH significance. This source of model uncertainty is related to the Level 1 / Level 2 Interface technical element discussed in Topic 2, Appendix D, Table A-4.

Potential R&D

This issue was identified as one that could potentially benefit from targeted dynamic PRA analysis.

6.6 Level 2 At-Power PRA Parameter Uncertainty

No parameter uncertainty issues were specifically identified. Methods for estimating parameters and characterizing their uncertainty are well established, and while the data may be sparse for some aspects of Level 2 At-Power PRA model development, the issues related to their estimation are not unique to the Level 2 PRA.

7. CONCLUSIONS

The purpose of the workshop was to gather experts together to gain a better understanding of the sources of uncertainty, how they are manifested in the PRA, and their potential significance to the PRA model and results. Therefore, the issues raised in each topical session were categorized as model uncertainty, completeness uncertainty, level of detail uncertainty, or parameter uncertainty. Priority was given to ascertain the significant sources of model uncertainty, as one of the workshop goals was to determine the uncertainty impact on the PRA model.

As the individual sessions discussed sources of PRA uncertainty, a subjective significance ranking was assigned to each issue of HIGH, MEDIUM or LOW as defined in Section 2. The total number of individual uncertainty issues raised in each topical session was as follows: 59 issues for internal fire, 22 issues for seismic events, 22 issues for low power and shutdown, and 30 issues for Level 2. It appears for seismic events and Level 2 that model uncertainty was the predominant source of uncertainty, while internal fire and low power and shutdown had a more even spread among the various sources of uncertainty.

Of the 133 total issues identified among all the topical sessions, 78 issues are expanded in greater detail in the body of this report. These topics were determined to be of MEDIUM or greater significance (occasionally a topic ranked LOW is discussed based on the discretion of the session facilitator). For each of these topics, the following information was presented: 1) a description of the issues or sources of uncertainty, 2) how the issues are manifested in the PRA is discussed, 3) a discussion of how the issues are relevant to the base PRA, application or both, if the issues are applicable to new, existing, or advanced reactors, and the significance ranking (HIGH, MEDIUM, or LOW) for that issue as related to the Standard or draft Standard technical element, and 4) a discussion of potential research and development (R&D) work which may be needed to resolve the issues or uncertainties. A final list of the number of HIGH, MEDIUM, and LOW issues raised in the four sessions compared to their uncertainty classification is presented below in Table 6.

Table 6. Summary of Workshop Session Rankings

Rank	# of Model Uncertainties	# of Completeness Uncertainty	# of Level of Detail Uncertainties	# of Parameter Uncertainties
Internal Fire (59 Issues Total)				
High	10	2	8	2
Medium	7	6	9	3
Low	5	1	4	2
Seismic Events (22 Issues Total)				
High	6	0	4	2
Medium	3	0	0	3
Low	4	0	0	0
Low Power and Shutdown (22 Issues Total)				
High	1	1	0	0
Medium	3	4	5	0
Low	2	3	1	2
Level 2 (30 Issues Total)				
High	11	1	1	0
Medium	12	0	0	0
Low	4	0	1	0

As seen in the above table, the LPSD topic area categorized significantly less issues as model uncertainty and concluded that the majority of the issues were less than HIGH significance. In comparison, the Level 2 topic area identified almost exclusively model uncertainty issues which were predominately assigned MEDIUM or HIGH significance.

A comprehensive list of each issue identified in the Internal Fire, Seismic Events, LPSD, and Level 2 sessions can be found in Appendices A, B, C and D, respectively. These appendices provide the source of uncertainty, discussion of the issue, type of uncertainty, significance, and possible resolution grouped by technical element for that topic area.

8. REFERENCES

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APPENDIX A - UNCERTAINTIES IN INTERNAL FIRE PRA

Name	Affiliation	Role	Presentation ADAMS Accession Number
Jeff LaChance	Sandia National Laboratories	Session Moderator	ML120680427
Ray Gallucci	Nuclear Regulatory Commission	Presenter	ML120680431
Paul Guymer	Jacobson Analytics	Presenter	ML120680439
Mike Wright	Jacobson Analytics	Presenter	ML120680439
Mardy Kazarians	Kazarians and Associates	Presenter	ML120680444
Dennis Henneke	General Electric – Hitachi	Presenter	ML120680442
Nick Melly	Nuclear Regulatory Commission	Note taker	-------------------

Table A- 1. Internal Fire PRA Sources of Uncertainty Grouped by Technical Element

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
Plant Boundary Definition and Partitioning (PP)				
1. Credited partitioning elements and fire barrier effectiveness.	Partitioning elements that **lack a fire-resistance rating, credit spatial separation or active fire barriers, introduce some modeling uncertainties.** The majority of these uncertainties are modeling preference.	Model	Low	None discussed
2. Credited partitioning elements and Fire barrier effectiveness.	Fire barriers are utilized to delineate the fire areas into physical analysis units. **Fire barriers failure rates** are included in the FPRA. Multi-Compartment Fires are typically low risk, but not always. No probability involved in the partitioning section, if there is any uncertainty it should be included in the multi-compartment analysis where fire barrier failure probabilities are utilized. The fire barrier uncertainty lies within the method of how penetrations failures are modeled when performing a multi-compartment analysis.	Parameter	Medium	Need clarification for accounting for multiple penetrations in a barrier
Fire PRA Equipment Selection (ES)				
3. Equipment selection of Multiple Spurious Operation (MSO)	NEI 00-01 includes a process for developing a plant specific MSO list, including review of the generic MSOs, and consideration for	Completeness	Medium	Need to examine potential for MSO causing release of radioactive waste (negative

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
scenarios is not complete.	plant-specific MSOs. **Incompleteness in the MSO selection process** can result in failure to identify risk-significant MSO scenarios. Likely the most risk-significant scenarios will be captured by other steps (e.g., modeling of the internal events PRA equipment).			impact on operator actions) – add to generic MSO list?
4. Equipment is selected that may cause an initiating event, including spurious operation.	Selection of possible fire-induced initiating events is incomplete, resulting in missing accident sequences. The uncertainty with this issue arises with selecting the initiating event caused by a fire. Knowledge of where cables associated with balance-of-plant systems is not always known and thus can require an assumption of which initiator the fire will cause. Uncertainty example: 1) Picking a bounding initiating event may not capture all initiating events in the room even if it is assumed to be the most severe. 2) Fire impacts can cause multiple initiating events at the same time	Level of Detail	Medium	None Discussed
5. Equipment (including cables) identified are mapped to the appropriate FPRA basic event.	It is common that all cables for each component are conservatively mapped to either multiple basic events (BEs) or the worst case BE. Typically, refinement of the cable selection process can limit the cables affecting each BE (typically performed on a case-by-case basis). However, the probability of each BE may still be conservatively estimated. For example, an	Level of Detail	Low	None Discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	MOV may be closed and spuriously close (driving the valve into the seat), thus preventing spurious opening.			
6. Equipment is selected that may cause a failure of a safe shutdown component, including spurious operation.	Equipment selection is not complete may result in an under prediction of risk. However, if equipment not selected is assumed failed in the Fire PRA, then the risk results will be conservative.	Level of Detail	High	None Discussed
7. Consider instrument air system (IAS) and the impact of a potential fire on the IAS.	Fire damage to instrument air lines can result in failure of the entire IAS given sufficient leakage. Soldered connections are looked at for a typical FPRA, which are assumed (if present) to fail the IAS. However, without soldered connections, the IAS is typically assumed unaffected by fire when IAS lines are in the fire area. Impact may be conservative (if IAS is not credited) or non-conservative (If IAS is credited for most fires). Most areas of importance are electrical, with minimal IAS lines. Additionally, fire damage to IAS lines is possible, but less likely than cable and equipment damage.	Model	Low	The uncertainty with this issue lies within the assumed guidance in NUREG/CR-6850 to fail instrument air when soldered connections are present. A review of this guidance is suggested. .
8. Equipment is selected that may impact the reliability of operator actions, including spurious operation.	Two Sources of Uncertainty: a) Equipment Selection is not complete, and may result in an under prediction of the risk due to no degradation in the HEPs impacted by failed instrumentation. b) Failure to identify potential undesired operator actions may result in an under	Level of Detail	Medium	None Discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	prediction of risk.			
Fire PRA Cable Selection (CS)				
9. Cables and circuits impacting selected equipment is identified and traced.	**Circuits and cables are not completely identified** and may result in an under prediction of risk. Typical reason may include a limit on the number of cables considered that may cause a spurious operation. Typically, the circuit analysis is performed using detailed and conservative safe shutdown procedures. Exclusion approaches (i.e., verifying cables are not in a location) are often used where cable routing is unknown.	Level of Detail	Medium	None Discussed
10. Cables and circuits impacting selected equipment is identified and traced.	**Assumed Cable Routing** may be inaccurate and may result in under or over prediction of risk, depending on the fire area. Assumed cable routing is typically performed for credited non-safety components, such as Main Feedwater and Condensate.	Level of Detail	High	None Discussed
11. Permissives, interlocks and associated logic are modeled in the FPRA.	Modeling quality can be impacted by either **conservative modeling (typical of safe shutdown analysis) or incomplete modeling of I&C.** Typically, the circuit analysis is performed using detailed and conservative safe	Level of Detail	Medium	None Discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	shutdown procedures.			
12. Electrical overcurrent protection is performed for credited power supplies.	Failure to perform **overcurrent protection analysis** can either credit potentially failed power supplies, or if the FPRA is conservatively modeled (power supplies not coordinated are assumed failed), the FPRA can be conservative. Typically the major power supplies are coordinated. In new FPRAs, non-safe shutdown equipment power supplies may not be analyzed.	Completeness	Medium	None Discussed
Qualitative Screening (QLS)				
13. Qualitative screening of fire areas not impacting the FPRA.	The above issues in ES may result in **fire areas** containing cables impacting the FPRA being **screened prior to quantification.** The addition of a few cables or components in a low-risk area will typically not result in the area becoming significant	Level of Detail	Low	None Discussed
Fire PRA Plant Response Model (PRM)				
14. Develop the FPRA response model to determine CDF and LERF.	The Fire PRA shall include the Fire PRA plant response model capable of supporting FPRA quantification. The **choice of FPRA PRM tools** may be a modeling preference, but it may result in some modeling and quantification limitations. The FPRA models also **inherit the**	Level of Detail	Medium	None Discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	limitations from the internal event PRA models (e.g., cannot quantify with TRUE runs, etc.)			
15. The PRM models the fire-induced initiating events or accident sequences.	**Over-simplification or failure to model new initiating events or accident sequences** can result in an under-prediction of risk. Given numerous new scenarios added as a result of MSOs, failure to model these scenarios accurately can result in significant errors.	Level of Detail	High	None Discussed
16. New Success Criteria should be developed and modeled.	**Use of existing success criteria** can result in either conservatism or non-conservatism, depending on the existing success criteria. The success criteria used in the internal events analysis (primarily timing) may change for a FPRA. However, success criteria (timing) is not drastically changed. New fire scenarios may require new timing requirements. In these cased new success criteria must be developed to encompass the specific event. In some cases new thermal hydraulics analysis will need to be performed.	Level of Detail	Medium	None Discussed
17. Modify the PRM to include new equipment, including spurious operations.	**Failure to model new equipment** may result in an underestimate of risk. Typically, there are a large number of modeling changes to support the FPRA. It is common that the modeling is complicated involving logic specific for the location of the fire.	Level of Detail	High	None Discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
18. Perform data analysis for new basic events.	**Incomplete data analysis** may result in conservatism or non-conservatism. Generally, the fire-damage impact of the new components is more important that the failure rate (e.g., events set to true when fire damage occurs).	Parameter	Low	None Discussed
19. Identify new accident sequences that go beyond CDF (e.g., impacting LERF).	**Failure to comprehensively review LERF sequences** may result in an underestimate of LERF. ES requirements include consideration for MSOs impacting ISLOCA. Additional accident sequences are possible and may be missed.	Completeness	Medium	None Discussed
Fire Scenario Selection and Analysis (FSS)				
20. Develop one or more fire scenarios (combination of ignition sources and targets) for each unscreened area such that risk is characterized or bounded.	A typical Fire PRA includes both: • Scenarios where the ignition source and target **grouping is conservatively performed,** resulting in conservative risk results. • **Incomplete scenario development** where not all risk-relevant combinations of ignition sources and targets are developed. FPRAs performed using NUREG/CR-6850 involves development and analysis of thousands of scenarios. Typically, most are conservatively modeled. For higher risk areas, it is possible to develop detailed scenarios where risk-relevant scenarios are not fully developed.	Level of Detail	High	None Discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
21. Select one or more scenarios for the Main Control Board (MCB) involving damage to more than one function.	**Simplified modeling of control room abandonment** scenarios may result in either conservatism or non-conservatism. **Failure to consider detailed fire-damage** which can potentially fail safe shutdown outside of the control room can result in non-conservatism. MCB scenarios are often risk significant. Plant knowledge and engineering judgment are needed to develop the detailed scenarios, without analyzing all possible scenarios. Typical FPRA modeling uses a bounding failure probability for control room abandonment, which may be conservative for most scenarios.	Model	High	None Discussed
22. The Fire PRA shall include an analysis of potential fire scenarios leading to the MCR abandonment.	**Simplified modeling of control room abandonment scenarios due to smoke** may result in either conservatism or non-conservatism. Typical FPRA modeling uses a bounding failure probability for control room abandonment, which may be conservative for most scenarios. Actual modeling of safe/alternate shutdown panels outside the Main Control Room is typically not done in FPRAs.	Model	High	NUREG/CR-6850 Appendix L spread and control model needs to be verified.
23. Analyze target damage times based on the thermal response of the target.	Fire Testing has shown that most cables can last for 30 minutes or more given a damaging fire. Without **consideration of thermal response** (cables are typically assumed to fail when the temperature reaches a specific value)**,** a Fire PRA can be a factor of 2 or more conservative. Cable	Model	High	Develop method for equipment damage

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	damage is also assumed in the FPRA when the cable tray is ignited, which may not be the case. However, there is no method presently developed to account for this last issue. A 20 minute additional time for suppression changes the risk by more than a factor of 5**.**			
24. Fire growth time is included in detailed fire scenarios	**Fire growth time** is often treated as a constant; 12 minutes for electrical fires, and 6 or 8 minutes for typical transient fires. However, growth time can vary, and may not be independent of heat release rate (HRR). Finally, some fires are assumed instantaneous (oil, hydrogen, etc.). Many Fire PRAs are dominated by electrical cabinet fires. Growth time of 12 minutes is likely conservative for most fires. High energy arc fires can result in instantaneous growth and are treated separately.	Model	High	More work needs to be performed to look at the correlation between growth rate and peak Heat release rate. Appendix E of NUREG/CR-6850 should be looked at and verified or updated. (Appendix E establishes the HRR curves)
25. If severity factors are applied, factors should be independent of other factors.	**Severity factors** are applied either as a result of fire modeling (minimum fire heat release rate to damage cable), or **using existing empirical or statistical models.** Fire modeling severity factors are generally based on conservative estimates (i.e., failure of the 1st target). Statistical and empirical models are based on generic models, and can be uncertain.	Model	High	New severity models have been developed but they need verification; additional model development is needed.
26. Establish and apply damage criteria.	Damage criteria are developed for generic types of cable or equipment, and may vary depending on the specific cables or	Model	Medium	None Discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	equipment installed. Other affects, such as cable loading, aging, installation of metal covers, and installation-specific factors can impact the time to damage for a specific cable. Variation within groups of cables is not large in comparison to variation among groups (e.g., thermoset versus thermoplastic).			
27. Establish and apply damage criteria.	Very small percent of thermoplastic cables may be excluded in the consideration of damage criteria. Significance varies. Plume scenarios within the zone of influence are not significant but hot gas layers are. Depending on the functions of these thermoplastic cables, generally this uncertainty has negligible impact on the final FPRA results.	Model	Low	None Discussed
28. If fire wraps are credited; provide a technical basis for rating.	Testing has shown variation in the ability of **fire wrap** to protect cables. Installation problems can result in wrap not protecting cable for the designed duration. Unlike barriers and penetrations, failure probabilities are not assigned to fire wraps. Typically, degraded fire wrap still provides sufficient protection to ensure the cables are low risk.	Model	Low	None Discussed
29. Apply fire modeling tools to account for fire growth, damage criteria and scenario	**Application of fire modeling tools** can result in either conservatism or non-conservatism.	Model	High	None Discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
specific attributes within the known limits of applicability.	NUREG-1824 provides guidance on the use of major fire modeling tools to various conditions and parameters. However, many of the entries are listed as "yellow" where "*there [are] calculated relative differences outside the experimental and model input Uncertainty.*" For example, all of the listed codes are listed as "yellow" for smoke concentration, which is one of the bases for the control room evacuation analysis.			
30. Type of fire propagation model.	Both intra- and inter-model uncertainty exist. Intra-model uncertainty addresses the variability that can be obtained if different models of the same type (e.g., zone or CFD) are compared. Inter-model uncertainty addresses use of different types (levels) of model, usually associated with greater and lesser degrees of refinement (e.g., more detailed modeling possible via a CFD model such as FDS vs. a zonal model such as CFAST). Tweaking the parameters has a high impact. The tools must be used in a correct manner.	Model	High	None Discussed
31. Fire growth resulting in propagation from one vertical cabinet to the next is included in the FPRA.	NUREG/CR-6850 includes deterministic rules on the **timing and spread of fires within cabinet groups** such as MCCs. A recent GE-Hitachi report shows fire growth between cabinets is unlikely and that the NUREG/CR-6850 may be wrong. Cabinet-to-cabinet fire growth is typically important due to the potential high HRR that results (not direct equipment damage). The	Model	Medium	Need better data to improve model

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	resulting large fire can be significant. However, most fires do not have to spread in order to damage and ignite overhead cables.			
32. Provide an assessment of smoke damage.	Evaluation of smoke damage is typically qualitative. Vulnerabilities are included in the quantitative model. Generally, the risk from **smoke damage** (other than for impact on human error probabilities) is considered low. However, it is possible that plant unique features could be vulnerable to smoke damage, and may not be captured by a qualitative review. Currently there is no modeling technique in FPRAs to quantitatively evaluate the impact from smoke damage. Generally, the risk from smoke damage (other than for impact on HEPs) is considered low based on observations from actual fires.	Model	Low	None Discussed
33. Manual suppression	Adjustments to manual suppression credit cannot be made to the fire brigade separately from the first responder. For example, a general model uncertainty issue is adjusting manual suppression for fire brigade response time. Currently, the entire manual suppression curve is adjusted. Also, need to evaluate potential for Fire Brigade actions leading to additional failures. Manual suppression can be applied to every fire scenario, but its model uncertainty is	Model	High	Need to develop method for adverse actions

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	unknown and potentially important. Long durations, which contribute to large non-suppression probabilities, can arise from responses by extinguishers alone. This runs counter to an easy assignment of importance with respect to model uncertainty. The fire database effort underway will provide the ability to better evaluate this issue.			
34. Estimate fire modeling parameters based on relevant generic industry and plant-specific information. Each parameter estimate shall be accompanied by a characterization of the uncertainty.	**Fire Modeling Parameter estimates** are typically either well known or applied as bounding estimates. This may include factors like room temperature, HVAC flow, wall material and thickness, etc. Generally low uncertainty in the parameters.	Parameter	High	None Discussed
35. The Fire PRA shall analyze scenarios with the potential for causing fire-induced failure of exposed structural steel.	Scenarios are typically analyzed only when there is **exposed structural steel** and a high hazard source located nearby. Plant risk for damage to exposed structural steel is generally low, except for selected plants (there have been occurrences in nuclear power plant fires).	Model	Medium	None Discussed
36. The Fire PRA shall evaluate the risk contribution of multi-compartment fire scenarios.	Initially, **multi-compartment analysis (MCA)** was considered low risk. However, some FPRAs are showing MCA scenarios in the risk-significant scenario list. Two factors appear to impact these results:	Model	Medium	None Discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	a) **Fire barrier penetration failures** in NUREG/CR-6850 are uncertain, and do not clearly state if this is for a single penetration or all penetrations on an existing barrier. Additionally, treatment of barrier failure given the fire source impact is not clear. b) **Conservative Fire modeling** for a single area results in conservative MCA results. Generally it is assumed that the combustible loading is spread throughout the fire compartment. Concentration of combustible material against a barrier could challenge the barrier.			
37. If exact cable routing is unknown, assume the cable is damaged.	It is not uncommon to not know specifically in a room where every cable is located. As a result, the **FPRA assumes the cable is damaged for every fire** until the cable is traced in detail. This has shown up as a major conservatism in several FPRAs.	Level of Detail	High	None Discussed
38. Select one or more scenarios for the Main Control Board involving damage to more than one function.	Incomplete scenario development where **not all risk-relevant combinations of ignition sources and targets are developed**. MCB scenarios are often risk significant. Plant knowledge and engineering judgment are needed to develop the detailed scenarios, without analyzing all possible scenarios.	Level of Detail	High	None Discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
39. The Fire PRA shall characterize the factors that will influence the timing and extent of fire damage for each combination of an ignition source and damage target sets.	**Realistic Estimates for Fire Damage is not performed:** Fire damage estimates typically start conservative, with more realism added to the top (risk-significant) scenarios. Details may include multiple heat release rate groups, inclusion of fire growth time, decay time, consideration for environmental conditions for realistic time to damage, and more detailed configuration considerations. Detailed fire modeling is time-consuming and is only performed for a limited set of significant scenarios. Realistic Fire Modeling for each scenario requires a significant effort. Typically, a majority of the scenarios are conservatively modeled.	Level of Detail	High	None Discussed
40. Use of generic fire modeling.	**Generic fire modeling** is often used to determine, for example, the minimal HRRs causing a damaging HGL, zone of influences for specific component types, etc. Application, other than possible ignition of intervening combustibles, is almost always conservative. Significant scenarios identified using generic fire modeling are further modeled using detailed fire modeling. Non-significant scenarios are typically not modeled further (e.g., are conservative).	Level of Detail	Low	None Discussed
41. Include an assessment of fire suppression effectiveness for each fire scenario	**Credit for Fire Suppression** is typically performed once the time to damage is determined from Fire Modeling. Generally speaking, with detailed fire modeling only performed on a small percentage of	Level of Detail	High	None Discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
being analyzed.	scenarios, the credit for suppression is conservative. Additionally, the existing NUREG/CR-6850 suppression curves are considered conservative (**no credit for control of fires** prior to suppression). Estimates of conservatism for suppression are a factor of 2 for a typical Fire PRA.			
42. Perform walkdowns on detailed scenarios.	**Walkdowns** are performed to confirm all of the modeled aspects of the scenario analysis. Generally, the walkdowns are performed to confirm and document modeled scenarios. Errors are possible given the amount of information collected.	Completeness	Low	None Discussed
Fire Ignition Frequency (IGN)				
43. Develop Fire Frequencies for each fire area and ignition source.	Difficult to identify **plant-specific "outliers"** due to rare events for a given component type.	Parameter	Low	None Discussed
44. Develop Fire Frequencies for each fire area and ignition source.	NUREG/CR-6850 supplement 1 is considered conservative with relation to the assigned **Heat Release Rates.**	Parameter	Medium	None Discussed
45. Develop Fire Frequencies for each fire area and ignition source.	Present NUREG/CR-6850 results in **different fire frequencies for the same equipment in different plants.** For example, older BWRs with less equipment than a new PWR may result in a factor of 2 higher fire frequencies for pumps or	Parameter	Medium	None Discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	electrical equipment.			
Quantitative Screening (QNS)				
46. Perform Quantitative Screening, including the establishment of screening criteria and verifying the impact to the FPRA results is small.	**Quantitative screening** is performed prior to applying all factors to ensure realism, based on relatively high screening criteria. Generally, the screening criterion does not greatly impact the final total CDF or risk significant basic events.	Level of Detail	Low	None Discussed
Circuit Failure Analysis (CF)				
47. Apply circuit failure (CF) probabilities for undesired spurious operations.	Existing **CF probabilities** range from 0.3 to 0.6, with an EF of around 2 to 3. However, method 2 in NUREG/CR-6850 results in lower results. The DC circuits expert panel is re-looking at these failure probabilities and will likely show significant changes from some events, especially method 2 or components with CPTs. Could result in a factor of 2 difference in the FPRA. DC Circuit Testing has shown some factors affect the CF probabilities.	Parameter	Medium	None Discussed
48. Apply circuit failure (CF) probabilities for undesired spurious operations.	NUREG/CR-6850 and other Fire PRA methods do not include the probability or approach for **considering Spurious Operation Duration for DC circuits**. This would include duration of spurious PORV, MSIV, and SRV openings. The average duration for DC Spurious Operations appears to be around 2-3 minutes.	Completeness	High	None Discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	For plants where MSOs contribute greatly to the overall risk, DC components typically are the most important. Short duration will mean the component will return to its failsafe position.			
Post-fire Human Reliability Analysis (HRA)				
49. Determine the time available and time to perform in support of detailed HRA	**Time-lines for HEPs** have uncertainty both on the time window for available time, based typically on T-H analysis and the time to perform, based on either simulator runs, walkdowns or talkthoughs. Fire HEPs add additional complexity, since the actions are typically in response to fire damage, which is typically conservatively estimated. Timelines for Fire HEPs are often times based on conservative estimates for fire-damage, and best estimate but uncertainty time windows for available versus performance times. There generally are no simulator exercises that cover fire procedures.	Model	High	Need better guidance for evaluating timelines
50. Identify new FPRA actions relevant to the FPRA PRM.	**New Actions include actions from the Fire Emergency Response Procedures**, as well as actions associated with new accident sequences. Some plants have area specific actions. A comprehensive review can be time-consuming. Generally, this is done completely, but a missed HEP can be significant	Completeness	Medium	None Discussed
51. Identify new	This action goes with the ES-C	Completeness	Medium	None Discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
undesired actions relevant to the FPRA PRM.	**identification of instrumentation potentially causing undesired operator actions.** Undesired operator actions are typically not significant.			
52. Model any existing or new FPRA actions including accident sequence specific factors (timing, etc.)	**Inclusion of the HEPs into the model** may include modification to an accident sequence, system model, or recovery of an event. Failure to properly model the HEP impact can result in either conservatism or non-conservatism. Generally, this is done completely, but a missed HEP can be significant.	Completeness	Medium	None Discussed
53. Perform Detailed HEP analysis for significant HEPs, including PSFs from Fire.	Results of **detailed HEP analyses**, especially when considering the fire-specific PSFs, are highly uncertain. Generally, most HEPs are lower risk. However, a few key HEPs are typically in the dominant sequences, such as control room evacuation. Estimates for detailed Fire HEPs are highly uncertain.	Parameter	High	None Discussed
54. Include operator recover actions that can restore function.	The **addition of recovery actions** is typically performed at the end of the FPRA. In addition to having high uncertainty for any recovery actions, Fire PRAs do not always credit recovery actions including procedural actions in the Fire emergency procedures. The total number of Fire PRA sequences makes the application of recovery actions difficult.	Completeness	High	None Discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	Estimates for detailed Fire HEPs are highly uncertain. Failure to include recovery values results in conservatism in the FPRA.			
Fire Risk Quantification (FQ)				
55. Model quantification shall determine that all identified dependencies are addressed appropriately.	**Dependencies in the HEPs** are common. Typically, the internal events methods are used, adjusting for the fire-specific HEP values. Estimates for detailed Fire HEPs are highly uncertain. However, the dependencies are less important than individual HEPs.	Level of Detail	Medium	None Discussed
56. The Fire PRA shall quantify LERF	**Attributes affecting LERF** are often times independent of fire effects. However, a limited amount of LERF contributors can be impacted by fire, which may not be accounted for in the FPRA modeling. As a result, LERF may be under predicted. It is not uncommon to fail to account for fire-impacts on LERF factors. For example, ISLOCAs may be more probable due to the potential for MSOs. Overall, the impact is moderate.	Level of Detail	Medium	None Discussed
57. Significant contributors to risk are identified.	Use of the **FPRA quantification t**ools (such as FRANC) can make it difficult to quantify either **importance measures or uncertainty values.** Work-arounds and add-on tools are being used to solve this problem. However, the process is not as robust as the internal events process. Work-arounds often times include a limited	Level of Detail	Medium	None Discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	amount of fire PRA sequence results.			
Seismic/Fire Interactions (SF)				
58. Qualitatively assess the potential for seismic/fire interaction issues in the Fire PRA.	The **SF assessment** looks at the impact of a seismic event on ignition sources, suppression and detection, plant response including brigade response, etc. The issue is treated **qualitatively** due to the estimation it is considered low risk in relation to seismic or fire risk analyzed independently. For some plants, the qualitative evaluation may miss vulnerabilities that are potentially significant.	Model	Medium	None Discussed
Uncertainty and Sensitivity Analyses (UNC)				
No Identified Issues				
New Area				
59. Potential for other hazards/fire (or fire/other hazard) interaction issues in the PRA.	The potential for multiple hazards (e.g., turbine blade ejection leading to fire and flooding) occurring should be investigated as is done with seismic-fire interactions. The issue probably could be treated qualitatively due to the estimation it is considered low risk in relation to hazards occurring independently. Consensus was that this should be analyzed in the hazard that causes the interaction. A qualitative evaluation may miss vulnerabilities that are potentially significant.	Model	Medium	Method/guidance is needed.

APPENDIX B - UNCERTAINTIES IN SEISMIC EVENTS PRA

Name	Affiliation	Role	Presentation ADAMS Accession Number
John Lehner	Brookhaven National Laboratory	Session Moderator	ML120680444
Annie Kammerer	Nuclear Regulatory Commission	Presenter	ML120680461
Jim Xu	Nuclear Regulatory Commission	Presenter	ML120680469
M. K. Ravindra	MKRavindra Consulting	Presenter	ML120680466
Greg Hardy	Simpson Gumpertz & Heger	Presenter	ML120680455
Michelle Gonzalez	Nuclear Regulatory Commission	Note taker	------------------
Brian Wagner	Nuclear Regulatory Commission	Note taker	------------------

Table A- 2. Seismic Events PRA Sources of Uncertainty Grouped by Technical Element

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
Probabilistic Seismic Hazard Analysis (SHA)				
1. Seismic source characterization.	The SSC model provides the characterization for all seismic sources that may impact a site of interest. Currently a number of specific technical questions related to SSC models are under discussion in the technical community. The workshop group did not get into these specific technical questions but felt that for seismic PRA SSC uncertainty was formerly high but was improving. The models used to characterize seismic sources from limited data have many uncertainties, but ultimately it is the output of these models that is used to determine an appropriate range of hazard parameters for the source characterization in the PRA model. So in the PRA this uncertainty manifests itself as a parameter uncertainty.	Parameter	Medium	The workshop group noted that this uncertainty is robustly and transparently captured in the PRA model by use of the SSHAC (Senior Seismic Hazard Analysis Committee) process (NUREG/CR-6372 and NUREG-2117). This uncertainty has been narrowed for some plants because of a new central and eastern US (CEUS) SSC model (NUREG-2115, EPRI 1021097), which was developed as a result of a new SSHAC Level 3 study conducted by the NRC, EPRI, and DOE.
2. Ground motion characterization.	Ground Motion Prediction Equations (GMPEs) provide a distribution of predicted ground motions for a particular magnitude and distance scenario earthquake. The GMC model incorporates a suite of appropriate and technically defensible Ground Motion Prediction Equation (GMPEs) into a Ground Motion Characterization (GMC) model using a logic tree framework. A host of specific technical questions related to GMPE and GMC	Parameter	High	There is a current effort underway to develop better data and GMPEs for the central and eastern US through the NGA-East (Next Generation Attenuation Relationships for Central and Eastern North America) project being jointly conducted by the NRC, EPRI, DOE and the USGS.

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	models are a matter of current expert discussion. A new GMC model is under development. The current model is generally hampered by the lack of data available at the time of its development. There is uncertainty in both the available data (or lack thereof) and the appropriate GMPEs to use. The workshop group did not get into the specific technical questions of GMC but felt that the uncertainty in the GMC models drive the uncertainty in PSHA analyses. The models used to characterize ground motion attenuation, based on limited data, have many uncertainties, but ultimately it is the output of these models that is used to determine an appropriate range of hazard parameters for the ground motion characterization in the PRA model. So in the PRA this uncertainty manifests itself as a parameter uncertainty.			NGA-East is a follow on to the successful NGA-West study that greatly enhanced GMPEs for the western US. The NGA-East project is being conducted using a SSHAC Level 3 process. As noted above, uncertainty is robustly and transparently captured by use of the SSHAC process, as described in NUREG/CR-6372 and NUREG-2117.
3. Site response: simplification and lack of standardization	Site response has significant uncertainty and a potentially large effect on the hazard results. However, site response techniques are not as standardized as they could be. Simplifying assumptions do not always apply and other tools are not well developed. Spatial and material variability is not always well captured and randomization approaches and tools are limited.	Model	High	The NRC is currently conducting research focused on addressing this issue.
4. Site response: lack of geotechnical information	Site response has significant uncertainty and a potentially large effect on the hazard results, but many operating plants lack geotechnical information from modern	Parameter	High	Obtaining better site-specific data, as a not very expensive option for improving on this

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	equipment for their sites.			uncertainty, was discussed for this issue during the session.
5. Spectral shape	Different approaches to developing the spectral input lead to different answers. Uncertainty in spectral shape arises from both the GMPEs and the use of a scenario earthquake or uniform hazard response spectra. The use of uniform hazard is usually conservative for design and for use in seismic PRA. Spectral shapes must be appropriate. They can be based on deaggregation or on a uniform hazard spectrum approach.	Model	High	None discussed
Seismic Fragility Evaluation (SFR)				
6. Soil-structure interaction (SSI)	Soil-structure interaction is very site specific. Soil-structure-interaction modeling is not well integrated with seismic hazard analysis or seismic PRA in terms of carrying through probabilistic loading.	Model	High	None discussed
7. Functional failure modes not clearly tied to the structural deformations	The relationship between the structure drift resulting from the seismic variable being used to describe the seismic hazard and the functional failure of the equipment attached to the structure is at best nebulous. Assumptions regarding the functional failure of the systems, structures and components relative to the seismic motion of the structure can significantly influence the PRA results.	Model	Medium	None discussed
8. Generic conversion of HCLPF to fragility	In some seismic PRA applications, the so-called hybrid method is used wherein the	Model	Medium to Low	None Discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	high confidence low probability of failure (HCLPF) capacity is calculated using the conservative deterministic failure margin (CDFM) method and the median capacity is estimated using a generic βc value. Using these parameters, the mean capacity and hence the mean fragility curve are approximated.			
9. Conservative assumption of structural failures	In the conduct of seismic PRAs usually conservative assumptions are made regarding structural failures of structures and components. This is done to make the analysis more efficient. For example, for the sake of efficiency the structure, system or component (SSC) is considered failed with the onset of yielding or buckling. Actually the SSC may be able to carry out its function beyond the point of yield or buckling. Conservative assumptions regarding structural failure may bias the PRA results, and may mask contributions of fragility of one SSC with regard to another.	Level of Detail	Medium	This uncertainty could be reduced with more detailed analyses of failure modes, but such an effort is likely to be quite costly.
10. Use of surrogate elements.	Attempts to capture the risk contribution via "surrogate" elements in seismic PRAs have not been very successful in the past. The ASME/ANS PRA Standard does not recommend their use. Analysts have rarely redone the core damage frequency calculations for different screening levels to assess the completeness issue. The use of surrogate elements mask potentially significant contributions of one or more systems, structures or components embedded in the surrogate element.	Level of Detail	Medium	None discussed but the obvious implication is not to use surrogate elements but rather a more detailed model (which will likely lead to a higher cost PRA).

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
11. Structure modeling	This issue was only briefly mentioned in the workshop but concerns the level of detail at which structures in a seismic PRA are modeled, for example in a simplified "stick" model or a more detailed finite element model.	Level of Detail	Medium	This issue was mentioned as being one that current or proposed research projects are attempting to address.
12. Sparse fragility test data	Test data is important in obtaining plant specific fragility. Complete fragility testing is rarely carried out. Usually a single qualification test is done and the failure level has to be extrapolated. Extrapolation models are used to predict system, structure and component behavior under beyond the testing range. This uncertainty may be even of HIGH importance for new reactors that have previously untested components with no, or very limited, fragility data.	Parameter	Medium	Additional fragility testing could reduce this uncertainty but would most likely be quite costly. Fragility tests are expensive.
13. Plant-specific loss of offsite power fragility	Loss of offsite power (LOOP) fragility is bound to be a significant contributor in a seismic PRA. Better plant specific analysis might remove unneeded conservatism in its estimate. Usually the fragility used is based on what it was done 30 years ago, and the data has not been updated. The LOOP fragility data should be more plant specific than it currently is. This uncertainty may be even of HIGH importance for new reactors that have previously untested components relevant for LOOP with no, or very limited, fragility data.	Parameter	Medium	Plant specific fragility could probably be improved with moderate cost.
14. Premature Screening of components	No discussion because of low significance due to the availability of methods to address this issue within the PRA.	Model	Low	None Discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
15. Success probabilities not fully considered	No discussion because of low significance due to the availability of methods to address this issue within the PRA.	Model	Low	None Discussed
16. Contribution from relay chatter effects not fully evaluated	No discussion because of low significance due to the availability of methods to address this issue within the PRA.	Model	Low	Research to address this issue is being undertaken jointly by EPRI and the NRC and the results are anticipated by the end of 2013. Both fragility data and guidance is being developed.
17. Only critical failure modes evaluated; contributions from other failure modes judged negligible	No discussion because of low significance due to the availability of methods to address this issue within the PRA.	Model	Low	None Discussed
Seismic Plant Response Analysis (SPR)				
18. Treatment of human error under seismic conditions.	The approach used for treating human error under seismic conditions is relatively crude. Human factors are not well characterized and may be very site specific. The human reliability analysis (HRA) models used in PRA can have significant influence on the results. A few actions can have a large impact in a seismic PRA.	Model	High	Improved human failure rate modeling for seismic conditions should be pursued. Suggestions included adapting fire HRA model methods with different stresses, using performance shaping factors that are used to analyze HRA in context of the scenario.
19. Treatment of correlation.	The treatment of correlation is usually "one fails-all fails," since the approach often taken in seismic PRA is to assume 100% response	Model	Medium to Low	None Discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	correlation as a starting point. If the issue of correlation then seems to make a difference to the overall results or insights, one can do a sensitivity analysis by assuming zero response correlation to ascertain how important the correlation might be, but sensitivity studies are often not thoroughly performed. This modeling uncertainty usually makes a difference for a few components (like diesel generators) but for most cases it does not lead to a big difference in results. However, it can be essential for some applications.			
20. Seismically-induced fire and flooding are not well developed or integrated.	Seismic induced fire and flood are usually treated in a qualitative manner in a seismic analysis. These items are disposed of usually via qualitative evaluation during walkdowns (and for floods, review of dams and ponds near the site); some quantitative studies have been performed. This model uncertainty was categorized as having unknown significance and was assigned a medium value by default.	Model	Medium (Unknown)	None Discussed
21. Simplification of the system model	Since many passive components and structures have to be included in a seismic PRA, for the sake of efficiency the seismic PRA plant response model usually starts with an internal events model that is simplified via various assumptions on initiating events and systems, structures and components (SSCs). this results in a simplified system model with a limited number of SSCs. The simplified model may miss potentially significant contributions of	Level of Detail	Medium to Low	None discussed but a more detailed model (which will likely lead to a higher cost PRA) would address this uncertainty.

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	one or more SSCs not modeled due to the simplification.			
Other				
22. Seismic PRA Updating.	Knowledge regarding seismic data and analysis techniques has evolved rapidly and significantly. There is uncertainty about the quality or viability of older seismic studies and the role of engineering judgment used. This can be an issue when a new analysis is used to update an old study, rather than to replace it. Specific guidance (based on guidance in the ANSE/ANS standard) is provided for situations in which an update should be performed. However, the quality of the technical basis of an older study is often a subjective decision.	Model	High	None Discussed

APPENDIX C - UNCERTAINTIES IN LOW POWER AND SHUTDOWN PRA

Name	**Affiliation**	**Role**	**Presentation ADAMS Accession Number**
Gareth Parry	ERIN engineering and Research Inc.	Session Moderator	ML120680475
Matt Dennis	Sandia National Laboratories	Session Moderator	ML120680475
Ken Kiper	NextEra Energy	Presenter	ML120680481
Don Wakefield	ABS Consulting	Presenter	ML120680489
Marie Pohida	Nuclear Regulatory Commission	Presenter	ML120680484
Steve Eide	Scientech	Presenter	ML120680479
Alysia Bone	Nuclear Regulatory Commission	Note taker	------------------

Table A- 3. Low Power and Shutdown PRA Sources of Uncertainty Grouped by Technical Element

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
Plant Operational State Definitions (LPOS)				
1. Omission of POSs needed to complete evolutions resulting from safe stable states from at-power scenarios	Some level 1 scenarios end in a safe-stable state, such as successful feed and bleed, successful shutdown to terminate SG tube leak, or sump recirculation following a LOCA. These may lead to prolonged shutdown to allow for repair. While they are low frequency scenarios, the complete cycle to restoration of power is not generally modeled. Associated with the characterization of shutdown POSs.	Completeness	Medium	None discussed
2. Level of refinement and characterization of POSs Note: This also is relevant for the development of the accident sequence models and the quantification	For time-averaged models, quantifications are performed once for each POS. If the plant condition value changes within a POS, the time assumed for determining the decay heat/RCS level/RCS temperature and pressure within each POS can impact the computed response times and success criteria. This is possibly less important when considering CDF averaged over many evolutions rather than for a specific outage. In addition, if the PRA is performed taking into account the requirements of the draft LPSD PRA standard, there is a requirement (LPOS-A6) ensures excessive conservatism is checked. Assumed Decay heat levels affects HEPs and success criteria although the analyst should define the POSs such that the values chosen should not make an HFE infeasible or change the number of trains required for system success.	Level of Detail	Low This was classified as low during the discussions, but depending on how rigorously a check for excessive conservatism is made, it could be of higher significance	None discussed
3. Use of the model	Future outage plans can and should be	Completeness	Low to Medium	None discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
developed from historical experience for future outages	reviewed as they may undergo changes with time. Accident sequence models can only assess known plans for future evolutions. The frequency of unplanned evolutions is problematical. Similarly, as the average durations change, they can affect initiator frequencies. PRA groups are mindful of these differences as overall outages shorten. Affects assumed durations of each POS and the times since plant trip which impacts decay heat levels and success criteria. Primarily classified as a PRA maintenance issue. Known changes can be factored into the model, and the ramifications in terms of POS duration, effect on initiator frequencies, assumed decay heat levels etc., accounted for. However, unknown and unknowable future changes are examples of sources of incompleteness.			
4. Selection of Outage Cause for Controlled Shutdowns and Forced Outages	Most LPSD models group forced outage evolutions by extent of RCS configuration changes required for repair rather than by a specific cause of outage. A representative cause of the outage type is then chosen. More severe causes, in terms of impact on mitigating systems, though low in frequency may be more risk significant. Typically the most frequent or common cause of each outage type is modeled as the cause of the outage; e.g. refueling, loss of main feedwater, RCS seal LOCA, or SGTR. Exceedance of an AOT caused by a more severe impact on a mitigating system (loss of an emergency AC bus) is not chosen.	Level of Detail	Low to Medium	This can be addressed by developing a greater number of representative outage types.

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
Initiating Events (LIE)				
5. Completeness of initiating events	Examples of initiating events that may be excluded include reactivity events other than boron dilution resulting from a loss of offsite power, and heavy load drops. These events are generally omitted because they are considered to be unlikely and could be candidates for a screening analysis.	Completeness	Low	None discussed
6. Availability and use of accident precursor data (example- drain down events not resulting in loss of RHR) from plants other than the one being studied.	The draft standard (for capability category II) currently requires a review of plant specific events for the identification of potential initiating events, but much more useful information may be available from industry data. For example, an event which did not cause an initiator at the plant at which it occurred may have done so at the specific plant analyzed due to differences in the plant evolution, plant design, or plant operational practices. Use of such data could be used to improve initiating event frequency data, by specializing the data to each plant and accounting for improvements in plant operations with time; e.g. adding additional level indication. However, this is contingent upon the availability of data and the level of detail that would allow such specialization. The specialization is likely to be a subjective process requiring assumptions to be made about the applicability of the data and its extrapolation. This can be categorized as a form of model uncertainty related to the interpretation of data. However, it would likely be manifested in the PRA model as a parameter uncertainty on initiating event frequencies.	Model	Low	The comprehensiveness of LPSD PRAs would be enhanced by the compilation of a data base with sufficient detail to allow the data to be reinterpreted for the target plant.

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
7. Grouping of Initiating events Note: The concern expressed here could also be classified under the LAS technical element.	Incorporation of phenomenological conditions (e.g. RCS break location, "bounding" break sizes, access to high temperature locations at < boiling), and debris (NPSH, plugging) into the sequence models for each POS, particularly for temporary conditions resulting from testing or maintenance, can vary with specific maintenance activities and alignments, LOCA size and LOCA locations. LOCA locations are typically not distinguished as separate initiating events for PWRs during at-power but it may be more important to do so during shutdown. During shutdown a large frequency contributor to LOCAs are maintenance actions inadvertently diverting flow from the RCS.	Level of Detail	Medium	None discussed
Accident Sequence Development (LAS)				
8. Modeling accident sequences by assuming that all failures occur at the time of demand may be non-conservative.	Assumption of operating equipment failing at time of first demand eliminates development of sequences for conditions after start; e.g. RHR relief valve is no longer isolated after pump start. Failure to credit RHR cooldown could lead to a similar omission. During SGTR, if RHR starts, subsequent RHR failure branches generally do not examine failures to isolate RHR allowing RCS depressurization through the RHR system following core uncovery; i.e. potential bypass scenario. Same sequence applies to shutdown conditions when initially on RHR even though not following a SGTR. This is not unique to LPSD but also applies to at-power modeling, and could lead to incompleteness in coverage of potential accident sequences.	Level of Detail	Medium	None discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
9. Modeling of repair/recovery potentially more important than for at-power models.	In at-power models, repair/recovery of a failed system is rarely credit, the exceptions being offsite power and diesel generators. However, for the shutdown scenarios where the options for success are typically fewer, and the time scale of the accident sequences may be longer (see #10 below), the modeling of repair or recovery may be more crucial. This could be either a model uncertainty (what model to use to estimate the probability of repair/recovery) or a parameter uncertainty if consensus can be reached on a model (e.g., the exponential model for recovery times). Because of the potential for recovery to be a significant factor, this is classified as medium.	Model or Parameter	Medium	None discussed
Success Criteria (LSC)				
10. Assumption of 24 hours as adequate for mission times.	24 hours is typically used as a default for at-power conditions. For both at-power and shutdown, some sequences may involve additional risk at later times due to sump plugging/ fuel assembly flow blockage concerns, for example. While this assumption is not unique to LPSD, it may have to be extended at least for some failure modes since failures after 24 hours may be significant. Random equipment failures after 24 hours are still not expected to be important. This is classified as a model uncertainty because there are issues related to how to define a safe, stable state. For example is continuing on sump recirculation or feed and bleed for an extensive time realistic. How	Model	Medium	None discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	long can it be assumed that RWST can be refilled? This is related to item #1 in LPOS. The model uncertainty, if resolved, would give an approach to resolving the level of detail issue addressed there.			
11. Applicability of computer codes and past generic analyses for shutdown sequence conditions; e.g. RCS vented with steam generators full.	There is an insufficient research base of SD scenarios to give us confidence that we are accurately characterizing SD success criteria. Lack of TH analysis results. Also, there is a question of the applicability of some codes. For example: can the codes analyze chugging effect. As a result, success criteria may instead be defined conservatively for selected conditions; e.g. no credit for SGs when RCS is vented regardless of vent size. Severe accident analyses for shutdown conditions would affect accident sequence development, HEPs, success criteria, and may impact severe accident event contributors. This could be classified as a completeness problem in that the knowledge base may not be large enough to cover all scenarios. There was some discussion that there is in principle no reason why some of the available codes cannot address the scenarios. Whenever a code is used, its limitations need to be recognized and reflected in the analysis. This may indeed lead to a conservative modeling in some cases.	Completeness	Medium	None discussed
12. Are there POS specific conditions under which systems cannot perform their required function?	The concern here is that while the systems may perform their function in most cases, there may be specific plant configurations where the system may not achieve its function. This may be related to issue # 11,	Level of Detail or Parameter	Low	None discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	but it may also be a result of not considering the spectrum of plant configurations adequately. This could be classified as a level of detail issue associated with the thoroughness with which plant configurations and their impact on systems is explored. It could also be considered a completeness issue as in #11 above.			
Systems Analysis (LSY)				
13. Identification of POS-specific system configurations.	There are specific configuration, spatial or environmental conditions that can affect system availability or long term reliability that may be different in different POSs. Examples include: temporary removal of flood barriers or fire barriers; reconfiguration of ventilation; instrument tube bolt detensioning with RCS not yet vented; NPSH concerns; plugging from debris in the shutdown following a LOCA); specific unusual system alignments. Identification of these conditions is more difficult than for at-power because of the many POSs and parallel activities going on. Such conditions may affect the feasibility of systems performing their function once an accident begins. Of particular concern are system conditions at the time of an RCS repressurization accident. The comprehensiveness of the coverage will depend largely on the skill of the analyst.	Completeness or Level of Detail	Low	None discussed
Human Reliability Analysis (LHR)				
14. Applicability of existing HRA methods for LPSD	There are significant differences between context for, and nature of, responses from those generally modeled for the at-power	Model	High	

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
conditions	scenarios. Examples include: The guidance available to operators in the form of procedures for the low power POSs vs. the shutdown POSs in that there is no equivalent to the EOP network for the latter; while there are abnormal procedures they don't have the same characteristics. For some responses more problem solving and skill-of-the-craft or knowledge based response planning may be required. Errors of commission can be more significant for initiating events. Some of the scenarios may be very long term scenarios, and thus repair and/or recovery of system functions can be more important Additionally, operator responses are relatively more important because many of the automatic means of responding to loss of a safety function are disabled. Since the methods that have been developed for at-power HRA are largely focused on procedure driven responses with limited requirement for diagnosis, the applicability of these methods to the LPSD, but particularly the SD POSs needs to be examined further. Also the HRA methods generally do not address repair or recovery, since these are typically handled using actuarial data. Specific issues that are identified as being unique to the modeling of LPSD include: • Treatment of dependency between at-initiator and post initiator HFEs (typically at power models don't			

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	address at-initiator HFEs with exception of those included in fault tree models for support system initiators) • Modeling of recovery and/or repair • Inclusion of specific errors of commission • Extendibility and applicability of at-power HRA models to SD conditions; is the PSF coverage and the guidance for assessing the effect adequate. • Additionally, there are issues that are relevant for at-power PRAs that may have an increased significance for shutdown conditions where there is increased reliance on manual actions. For example, should there be a cutoff value for multiple HEPs in an accident sequence cut set, and if so, what should it be? Should the cut-off value be variable depending on the context? This is particularly challenging if the time available for response is protracted.			
15. Criteria used for feasibility analysis	As an example, one of the inputs to assessing the feasibility of an operator action is how to assess the reasonableness of access to high temperature locations containing near boiling water. Time to boiling is often used as a limit to determine the time for access, but confined spaces may require much lower temperature limits in practice.	Model	Low	None discussed
16. Identification of at-initiator HFEs	The issue is a concern about the completeness in identifying at-initiator HFEs via reviews of industry operating experience and related reviews of plant specific test and	Level of Detail or Completeness	Medium	None discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	maintenance activities as part of the pre-initiator HFE evaluation process. The number of procedures available for review is huge and the search criteria for identifying such HFEs are not well established. Historical records are substantial but not always sufficiently documented to extrapolate their applicability to other plants. Further use of a pre-defined set of initiators dissuades analysts from examining individual causes within the defined and thereby account for plant specific unique boundary categories conditions. It is to some extent a level of detail issue that depends on the rigor with which an analyst performs the search. However, it could be classified as a completeness issue. Medium significance because of the possibility of significant plant to plant variation in system configurations and maintenance practices.	.		
Data Analysis (LDA)				
17. POS specific parameter estimates	Available digested data sources for loss of RHR are limited and raw event summaries are sketchy. Tendency is to use at-power failure rates for most equipment other than for RHR pumps. The influence of shutdown activities is known to be substantial on the loss of offsite power frequency and the concern is that other parameter failure rates may also be similarly affected during shutdown. Testing is more frequent/extensive during shutdown potentially leading to greater chances of detection and hence frequencies	Level of Detail or Parameter	Low	None discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	per unit time of corrective maintenance. Maintenance durations are not as constrained by tech specs. However, the timing of the testing is very likely at the most opportune time when availability of the equipment is not crucial. If POS-specific conditions are known to affect parameters, such as unavailability due to maintenance, then differentiating estimates between POSs is appropriate. If this were done, since it is likely that the data available is relatively sparse the parameter uncertainties will be larger than for cases where the data base is more extensive.			
18. Applicability of CCF parameter estimates derived from generic data for at-power conditions for use for specific plants during shutdown.	Generic CCF parameter data is generally screened for applicability to at-power conditions but not specialized to a specific plant as in NUREG/CR-5485. The plant specific determination is likely to be more important for shutdown conditions since many events result from maintenance. Two points were discussed: (1) Plant-specific specialization of the data generally requires making many assumptions about the applicability of the data to the specific plant. This is not required for CC II. There are uncertainties associated with the specialization but it can be captured as a parameter uncertainty. (2) The way the CCF events were analyzed for INL database was questioned. Specifically was consideration of LPSD context taken into account? maybe suggestion to NRC to visit this to see if there is a difference among CCF between at power and LPSD; right now, looking at all reported events.-- some	Model or Parameter	Low	The question concerning the interpretation of the raw data for the INL database is unresolved and could be topic for research.

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	disagreement on this topic. Whether plant-specific specialization of the raw CCF data is performed is a question concerning the state-of-practice, but if it is done, there are model uncertainties associated with interpretation of the data but they could be reflected as parameter uncertainties.			
Quantification (LQU)				
19. No issues unique to LPSD	As an example, one concern raised was the validity of the assumption of convergence when a one decade decrease in truncation limit changes CDF and LERF by less than 5%. "This is the same criterion as for at-power, and is especially of concern when post-processing of cutsets is used. Depending on the application, this convergence may not be sufficient; e.g. ranking of risk significant components. For shutdown conditions, this approach is less convincing since the problem is divided into many POSs; i.e. the problem is further fractured. Further, reliance on operator actions for sequence mitigation is greater during shutdown making the post-processing of cutsets that much more important. On the other hand, the length of accident sequences for shutdown is likely shorter than for at-power conditions meaning that equivalent truncation limits may be convergent after all." This is not strictly speaking an uncertainty concern and can be dealt with by taking care to consider dependency and post-processing when assessing convergence.	N/A	Low	None discussed
LERF (LLE)				

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
20. Omission of Potential LERF contributors: - Hydrogen combustion (equipment survivability), - steam explosions (RV head removed), and - induced RHR system failure (containment bypass)	These three potential contributors are excluded from the current list to be considered in Table 3.2.8-3 of the draft LPSD PRA standard (for PWRs). Relating to the last bullet, the potential for RCS pressure boundary failure following a loss of all cooling may be of interest. In POSs when the RCS is still intact but is at lower initial pressure, the SGs may also be at atmospheric pressure and the lower decay heat present may mean that during subsequent RCS heatup, that a more uniform set of RCS temperatures occur after core uncovery that increases the potential for RCS pressure boundary failure relative to that seen for induced SG tube ruptures initiating from an accident initially at-power. RHR RV's failing open as the RCS pressurizes could lead to rapid overheating of the RHR system after core uncovery. These contributors would affect the sequence development and contributors to LERF.	Completeness	Medium	None discussed
21. Availability of computer codes and past generic analyses for shutdown sequence conditions: - Source terms for shutdown conditions - shutdown on RHR cooling with RCS pressurized (induced SGTR), - RCS depressurized with RV head unbolted/removed	Past generic analyses to address these issues are not believed available. Can affect contributors to LERF. This is a concern about the completeness of the knowledge base. Making specific assumptions to deal with this lack of knowledge would lead to a model uncertainty.	Completeness or Model	High	None discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
and obstructions between RV and containment dome are removed (steam explosions).				
22. Assumptions regarding operator actions guided by procedures when the guidance is left to a decision of the Technical Support Center (e.g. several accident management guidelines and security-related mitigation measures) addressing known trade-offs between recovery event impacts; e.g. recovery and restart of containment spray)	PRA models may exclude such actions or assume they will be performed only when helpful. Inclusion of such actions when the actions mistakenly make things worse has not been included as an alternative assumption. HRA models developed for at-power level 1 PRAs do not address decision-making absent clear procedural guidance. Additionally, Severe Accident Management Guidelines (SAMGs) and Extreme Damage Mitigation Guidelines (EDMGs) were developed with an event from at-power in mind. These guidance documents may not fit some shutdown POSs well.	Model	Medium	None discussed
23. Assumption that quickly closed containment hatches (without fully bolting) have the same overpressure capacity as initially fully closed hatches	Equipment hatch closures often only require 4 bolts to be tensioned. Capacity analyses for such conditions have not been performed.	Completeness	Low	None discussed

APPENDIX D - UNCERTAINTIES IN LEVEL 2 AT-POWER PRA

Name	Affiliation	Role	Presentation ADAMS Accession Number
Don Vanover	ERIN Engineering and Research, Inc.	Session Moderator	ML120680495
Tim Wheeler	Sandia National Laboratories	Session Moderator	ML120680495
Don Helton	Nuclear Regulatory Commission	Presenter	ML120680509
Dr. Richard Denning	The Ohio State University	Presenter	ML120680498
Mark Leonard	dycoda LLC	Presenter	ML120680515
Jeff Gabor	ERIN Engineering and Research, Inc.	Presenter	ML120680501
Ray Schneider	Westinghouse Electric Corporation	Presenter	ML120680518
Sandra Lai	Nuclear Regulatory Commission	Note taker	-------------------

Table A- 4. Level 2 PRA Sources of Uncertainty Grouped by Technical Element

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
Level 1/Level 2 Interface (L1)				
1. Treatment of dependencies across the Level 1/Level 2 interface	Many approaches to transferring information across the interface between the Level 1 and Level 2 portions of the PRA are being used. These approaches include grouping sequences into core damage (or plant damage) states and direct transfer of sequence cutsets across the interface. Treatment of the following categories of dependencies across the interface have been identified as being important and will help to address how dependencies are treated between Level 1 and Level 2 model. • Initiator and support system dependencies • Prior equipment failures • Operator action dependencies (including available time) • Functional dependencies (including degraded plant conditions, e.g. RCP seal failure impacts on TI-SGTR) • Common cause dependencies • Treatment of off-site power recovery on late accident progression and mitigation	Level of Detail	Low	Proper treatment of these issues is encompassed within the Level 2 PRA Standard.
2. Number of plant damage state (PDS) groups	Grouping accident sequences or cutsets from the Level 1 PRA into plant damage states for input into the Level 2 PRA potentially introduces uncertainties due to the resulting loss of modeling detail. Care should be taken to ensure that information is not lost due to PDS simplifications. The following questions should be considered. • Are the number of PDS groups sufficient to represent the significant differences among the Level 1 sequences? • If fewer PDS groups are used, does the	Level of Detail	High	Proper treatment of these issues for the base model is encompassed within the Level 2 PRA Standard.

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	"representative" sequence reasonably bound the set of sequences assigned to the PDS and are the intergroup sequence characteristics sufficiently similar such that the representative sequence does not create a overly conservative or non-conservative bias in the modeling?			
3. Partial / degraded performance not credited in Level 1	Numerous Level 1 PRA modeling choices can be influenced by the go / no-go nature of Level 1 PRA end-states. In some cases, partial flow from systems or injection flow from lower capacity systems not credited in the Level 1 PRA model can have an adverse impact on the severe accident progression. For instance, injection of water into a degraded core might be able to prevent vessel failure, but there is also the potential for increased fuel–coolant interactions leading to additional releases of hydrogen and fission products.	Completeness	High	Focused investigations may be warranted
Containment Capacity Analysis (CP)				
4. Core debris containment boundary failure	Direct contact of core debris with the containment is a significant containment challenge that may lead to early containment failure. Drywell liner melt-through has been found to be an important contributor to early containment failure for Mark I containments. An assessment of the probability of Mark I containment failure by core debris of the liner were published in NUREG/CR-5423 and NUREG/CR-6025. These studies indicated that in the presence of an overlying water pool the probability of early containment failure by melt-attack would be very low (order of 1E-03). However, for sequences with a dry pedestal region the probability of containment liner melt- through	Model	Low	General consensus appears to have been achieved on this issue.

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	appears to be relatively high (nearly 1).			
5. Primary containment structural vulnerabilities	The results of individual plant examinations indicate that specific containment features may lead to unique and significant failure modes.	Model	Low	The major vulnerabilities have been identified for the different types of containment. However, the potential exists for unrecognized vulnerabilities due to construction or design faults.
6. Containment failure modes given quasi-static loads	The mode and location of containment leakage and failure is one of the most important parameters impacting the magnitude and timing of radionuclide release. Multiple approaches have been taken to assessing containment failure mode and location including use of failure information from similar plants and detailed structural analyses. The analysis also needs to account for material creep/degradation due to high temperatures.	Model	Medium	Different modes of containment failure can typically be factored into the CET structure.
7. Dynamic load impacts on containment failure mode	Severe accidents can lead to environment conditions beyond those considered during the design of the containment system. Containment failure mechanisms caused by (or influenced by) accident phenomena/conditions such as the following should be considered: • hydrogen combustion (deflagration and detonation) • hydrodynamic loads • interactions between molten core debris and water The containment response is highly dependent on the geometry and definition of the impulse.	Model	High	These issues are typically handled with separate engineering analysis. In some cases, a bounding treatment may be sufficient to show that the probability of failure is low. If this is not the case, then the potential to become an important source of model uncertainty increases.

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
8. Indirect mechanisms of containment failure	Severe accident phenomenon may lead to containing integrity challenges in addition to high static or dynamic pressures. Mechanisms such as those discussed below may also challenge containment integrity. • Debris concrete interactions have the potential to result in reactor cavity/pedestal structural failure. • RPV lower head failure under high pressure conditions may result in reactor cavity/pedestal structural failure. • Ex-vessel steam explosion may potentially cause reactor cavity/pedestal structural failure. • Seismic induced leakage may occur (e.g., through penetrations) for well-beyond design basis earthquakes.	Model	Medium	Depending on the basis that is established for each of these issues and the associated importance measures for specific applications, sensitivity cases may be warranted to examine the potential impacts from alternate assumptions (i.e., different failure likelihoods) associated with each of these issues.
9. Quasi-steady failure threshold methods and correlation between failure pressure and leak rate	The ability to determine failure pressure given defined conditions has significant uncertainty (greater for concrete containments). Significant uncertainties are also associated with construction detail and ageing effects. The basis for developing fragility curve is typically subjective. Ultimately, the containment failure capacity is characterized by a point estimate (e.g., lower bound or “best” estimate pressure) or by a probability density function (fragility curve).	Model	Medium	None discussed
10. Containment failure characteristics	Given containment failure occurs, the source of uncertainty relates to how the “final” containment failure is characterized (i.e., location and size). The containment failure size could be a function of containment load (e.g. pressure) or time (e.g., debris liner contact). The containment failure could also be characterized by a ‘threshold’ model or a ‘leak before break’ model.	Model	Medium	Different containment failure characteristics can typically be factored into the CET structure.

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	Is the containment failure characterized by a 'threshold' model or a 'leak before break' model? The threshold model defines a threshold pressure at which the containment is expected to fail with a large breach. In the leak before break model, containment leakage is expected to precede a major rupture and the leakage rate is modeled to increase with increasing internal containment pressure up to the ultimate capability pressure, at which point a larger failure of the containment is expected to occur. If the rate of addition of mass and energy to the containment atmosphere is smaller than or equal to the leakage rate, further containment pressurization is not expected to occur and catastrophic failure of the containment may be averted.			
Severe Accident Progression Analysis (SA)				
11. Thermally induced failure of RCS pressure boundary (PWRs)	For PWRS, this issue is associated with thermally induced failures under high pressure conditions of hot leg piping/vessel nozzles, surge lines or steam generator tubes. The probability of a thermally induced rupture of steam generator tubes depends on several factors including: • Treatment of natural circulation and loop seal clearing • The thermal-hydraulic conditions (temperature and pressure) in the RCS and steam generators • Material properties impacting creep rupture • Presence of defects in the steam generator tubes Large amount of available information to support failure likelihoods for existing PWR fleet. May be a	Model	Medium	None discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	potentially larger source of uncertainty for unanalyzed reactor designs.			
12. Thermally induced failure of RCS pressure boundary (BWRs)	For BWRs, the severe accident progression can also result in thermal induced failures of the pressure boundary. Failure of the RCS can lead to RPV depressurization prior to vessel breach, and depending on the failure location can have a significant impact on fission product transport and release. Key issues to consider include: • Treatment of the SRV stochastic failure probability due to cycling demands at high RPV pressure (a stuck open SRV would depressurize the RPV and lead to fission product transport to the suppression pool) • Material properties impacting creep rupture of the main steam line (main steam line failure would lead to bypass of the suppression pool) The timing of a stuck open relief valve versus continued heatup of the main steam line leading to failure can have a significant impact on fission product transport and release.	Model	High	None discussed
13. Recovery of a degraded core	Phenomenological issues associated with in-vessel core melt progression and retention are highly complex and uncertain. Important issues include: • Cladding oxidation behavior • Fuel and clad melting and relocation mechanisms • Crust formation/crust failure in the lower portions of the fuel These issues impact the potential for recovery of a degraded core. In many Level 2 PRAs, credit for in-vessel accident mitigation has been modeled for	Model	High	The impacts from recovery of a damaged core could impact the development of appropriate accident management strategies. Focused sensitivity studies and additional research might be warranted.

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	sequences where water flow was restored within a short period of time of the onset of core damage and prior to significant core geometry changes.			
14. RPV lower head failure mechanism	Alternative lower head failure mechanisms should be considered such as: • Global (creep) failure of reactor pressure vessel • Local failure of lower head of reactor pressure vessel (e.g. at lower head penetrations) • Early RPV leakage via failed open instrument tubes (also leading to a potential bypass of containment)	Model	Medium	Different RPV failure mechanisms can be factored into the CET structure.
15. External cooling of RPV lower head	The conditions associated with a molten pool in the lower head region are very uncertain. The ability to model side failure, unzipping, localized attack, or penetration failure depend on nature of the pool or debris. For some plants, there is uncertainty as to whether the vessel can be cooled externally. The issues to assess include: (a) whether the imposed heat flux exceeds the heat removal capability (critical heat flux) on the external surface (b) the potential for melting of the vessel wall under the thermal loading from the molten pool, and (c) the pressure bearing capability of the vessel wall held at high temperature inside, and low temperature outside. Potential High Source of Model Uncertainty for those designs that credit this means of averting vessel failure.	Model	High	None discussed
16. In-vessel fuel coolant interactions (steam explosions)	Calculated in-vessel loads resulting from fuel coolant interaction are considered to be well below the capacity of the reactor pressure vessel and are currently considered not to be a viable threat to reactor vessel or containment integrity.	Model	Low	None discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	General consensus appears to have been achieved on this issue.			
17. Ex-vessel fuel coolant interactions (steam explosions)	In contrast to in-vessel FCIs the calculated loads for ex-vessel FCI events may exceed the structural capacity of a typical cavity (i.e., sub-cooled water, lower pressure, weaker structure than the vessel). Ex-vessel FCIs may also impact accident progression and fission product release by: • Debris transport outside of cavity and/or pedestal • Enhanced hydrogen production • Releases of radioactive material Unclear potential effect of structural failures in cavity/pedestal region on containment integrity. Could have beneficial effect related to debris coolability and reduced core-concrete attack. Plant-specific susceptibility to this issue could affect the SAMG strategy.	Model	High	None discussed
18. Direct containment heating / high pressure melt ejection	Although high pressure melt ejection may not directly challenge containment integrity for many containment designs HPME may influence the accident progression and fission product release due to: • Additional heat generation and hydrogen production from zirconium oxidation in the steam/air containment atmosphere • Debris transport outside of cavity and/or pedestal • Initiation of hydrogen combustion • Enhanced releases of radioactive material Other than the ancillary debris spread issues, general consensus appears to have been achieved on this issue.	Model	Low	None discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
19. In-vessel hydrogen generation	The extent of in-vessel hydrogen generation is believed to be sensitive to a number of parameters including: • The extent of in-core flow blockages core • Cladding ballooning • Recovery and addition of water • Relocation of molten fuel • Zr melt breakout temperature • Fuel rod collapse temperature • Fractional local dissolution of UO_2 in molten Zr • Melt relocation heat transfer coefficient • Particulate debris characteristic size following core collapse • Particulate debris characteristic size following relocation to lower plenum • Porosity of fuel debris beds These issues are typically addressed via the code used for accident sequence progression analysis (e.g., MAAP or MELCOR). However, the uncertainty arises in the actual amount of total hydrogen that is generated for each sequence type and how that is factored into the Level 2 PRA model.	Model	Medium	None discussed
20. Energetic burning of hydrogen and combustible gases	The quasi-static and dynamic loads imposed on the containment structure as a result of hydrogen and combustible gas burns is impacted by a number of factors including: • Mixing and/or stratification of the containment atmosphere • Extent of steam inerting • Propagation of ignition and deflagration flames • Flame acceleration and transition from deflagration to detonation	Model	High	None discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	• Ignition sources • Heat losses to structures Good understanding of flammability limits, thresholds for deflagration and detonation for hydrogen. Less understanding of combination of hydrogen and CO. Limited understanding of conditions resulting in transition to detonation.			
21. Ex-vessel debris bed coolability	The coolability of core debris relocated to the reactor cavity/pedestal regions is subject to a number of uncertainties. One of the most important is the effective upward heat flux to an overlying water pool. A critical question is whether water penetration through the upper debris bed surface (e.g., through cracks) will facilitate heat transfer at rates above conduction limited heat transfer through a solid crust. There are also still substantial uncertainties in the two-dimensional cavity erosion profiles (i.e., heat flux partitioning between axial and radial directions). Note that excessive radial erosion can undermine containment integrity, while excessive axial erosion can fail the basemat, leading to ground contamination and release of radiological source terms in the environment. If debris bed is not coolable, then there is potential for a large impact on magnitude and type of late releases and land contamination issues.	Model	Medium	None discussed
22. Impact of core debris / concrete interactions	Core debris concrete attack can result in: • Undermining of containment structures (cavity walls/vessel support) by the core debris • Generation of non-condensable gas (H_2/CO/CO_2)	Model	High	None discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	• Lateral spreading of debris and potential for contact with containment pressure boundary • Potential for groundwater / environmental releases of radioactive material Potential large impact on magnitude and type of late releases and land contamination issues.			
23. Ex-vessel hydrogen and combustible gas generation	The extent of ex-vessel hydrogen and combustible gas production during core concrete interactions is impacted by a number of uncertain parameters including: • Ex-vessel debris/water heat transfer parameters • Enhancements to upward heat transfer by penetration of overlying water into cracks and fissures in the debris crust • The extent of sideways versus downwards concrete erosion • Concrete aggregate material composition • Quantity of remaining metals in the melt (zirconium and steel) These issues are typically addressed via the code used for accident sequence progression analysis (e.g., MAAP or MELCOR). However, the uncertainty arises in the actual amount of total combustible gas generation that occurs for each sequence type and how that is factored into the Level 2 PRA model.	Model	Medium	None discussed
Probabilistic Treatment (PT)				
24. Modeling of operator actions during severe accidents	The human error probabilities should be developed using a methodology that is consistent with the Level 1 PRA analysis. However, there are unique performance shaping factors (PSFs) that should be considered in the development of the Level 2	Model Human Reliability Analysis is recognized as a generic source of	High	None discussed

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	operator actions. While some Level 2 HRA PSF-equivalents are very decision-specific, some scenario-specific aspects exist: • Reluctance to make any decision that directly results in a release • Communication and decision-making between the control room, technical support center (TSC), and emergency operations facility (EOF) • Parsing of failed versus reliable indication Human Reliability Analysis is recognized as a generic source of potential model uncertainty in applications.	potential model uncertainty in applications.		
25. Treatment of SAMG (and other accident management) actions	This affects HRA and accident progression analysis portions of the Level 2 PRA. Most SAMG actions inherently have a positive and negative effect (e.g., containment sprays reduce containment pressure but increases likelihood of a hydrogen deflagration) As such, focusing only on "important" post-core damage operator actions may not be sufficient if the goal is to be best-estimate	Model	High	Similar to Item #3, focused investigations may be warranted.
26. Equipment / instrument survivability for SAMG implementation	This issue affects both explicit (e.g., equipment availabilities) and implicit (e.g., assumptions about available indication and its effects on operator response) pieces of the Level 2 PRA. The Level 2 PRA model assessment would need to consider not only pressure, temperature, humidity, and radiation impacts, but also the potential effects of hydrogen transport and deflagration/detonation into the reactor building or auxiliary building.	Model	High	Some credit for systems under severely degraded conditions could improve risk profile and realism.
27. Random and/or seismically induced failure probabilities	Unlike Level 1 PRA, equipment used in the severe accident management guidelines (SAMGs) often does not have the necessary data to support data-	Model .	Medium	This issue should improve over time as equipment credited in

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
not covered in Level 1 PRA data collection	informed failure probability assignment.			the Level 2 analysis becomes more main stream.
28. Correlation introduced by common physical parameters	NUREG-1855 discusses one type of correlation, the state-of-knowledge correlation (SOKC) or epistemic correlation which arises when the same parameter uncertainty model is used to quantify the probabilities of two or more basic events. Another type of correlation relates to phenomenological events which are correlated through dependencies on other common causal events/parameters. For example, in-vessel radionuclide release and hydrogen generation are not independent but correlated through dependencies on common accident progression parameters.	Model	Medium	This issue should not be important for models which are relying on the use of point estimate mean values, but a method for including these dependencies appropriately in a parametric uncertainty analysis has not been defined.
29. Passive system reliability	Some of the new reactor designs are relying on passive features to mitigate and/or reduce the impacts of a potential severe reactor accident. Definitive knowledge about the reliability of these systems and how that is factored into the Level 2 PRA model development process may not be well established.	Model	High	Potential to reduce uncertainty with focused research and experiments.
Source Term Analysis (ST)				
30. Source term characteristics	In addition to uncertainties introduced by uncertainties in the accident progression phenomena additional uncertainties exist for radionuclide formation, transport and deposition related to: • In-vessel fission product release • Fission product retention in the RCS and containment • Fission product chemistry • Fission product release during core debris concrete interactions	Model	Medium	The understanding of fission product behavior is improving but there are still significant uncertainties. This is more of a long term health effect issue for Level 3 analysis.

Source of Uncertainty	Discussion	Type of Uncertainty	Significance	Possible Resolution
	• Late revolatization from the RCS and containment • Fission product scrubbing in water pools • Fission product revolatization from water pools • Fission product grouping			
31. Source term attenuation in structures outside the primary containment	Secondary containment/auxiliary building may represent an additional effective retention area for radionuclides for certain types of sequences or containment leakage failure modes. For example prior PRAs have credited auxiliary/safeguards buildings for fission product attenuation for ISLOCA containment bypass sequences. Uncertainties arise as a result of the structural capacities of these structures (many have blowout panels, low pressure ducting, etc.), the impacts of potential phenomenological events in these structures (e.g. hydrogen burns) and the assessment of the release pathways from these structures. Some impacts on short term releases, but more important for long term health effects.	Model	Medium	None discussed
32. Impact from accident duration truncation of sequence runs	Truncating deterministic accident progression simulations (e.g., terminate calculation at 48 hours) could non-conservatively bias results toward risk from earlier releases. Assumption that recovery actions are 100% effective after some time may not provide the best estimate presentation of the results. Some impacts on short term releases, but more important for long term health effects.	Model	Medium	None discussed