Scoping Study on Advancing Modeling Techniques for Level 2/3 PRA

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This document does not represent a consensus of any specific group of individuals, but has benefited greatly from viewpoints expressed by the following NRC individuals:

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EXECUTIVE SUMMARY

This document is the first step in a scoping study to explore advancement of the methodology used for Level 2 and Level 3 PRA modeling. It provides a scrutable basis for the formulation of the work scope for a subsequent contractor-led effort. Primarily this report relies on existing technologies (including simulation-based analysis tools (e.g., MELCOR)) and explores how these tools can be used to arrive at more rigorous quantification of severe accident risk. The intent is not to affect the near-term treatment of Level 2 PRA in regulatory practices.

A number of past studies (e.g., NUREG-1150) are reviewed for the purpose of characterizing the similarities and differences in the approaches employed. Also, a synopsis of Level 2/3 regulatory considerations is provided, including discussion of the relevant regulatory requirements, as well as the way that Level 2 and Level 3 models appear in the agency's PRA tools. Next, a review is performed of advances that have been made in academia or elsewhere, but that are not reflected in the common (domestic) practice for Level 2/3 PRA.

Following this, feedback on Level 2/3 PRA methodology issues received from external stakeholders is summarized. Next, four general Level 2/3 approach categories are outlined, for the purpose of allowing the comparison of an otherwise unlimited number of possible approaches. The four general categories are:

- a modified traditional approach,
- a hybrid approach using system code-based event tree development,
- a dynamic event tree simulation approach, and
- a sampling-based simulation approach.

One perspective on how these approaches represent methods evolution can be found in the figure below.

Next, a specific approach category is selected for further exploration. Following this, possible variations in the selected approach category are explored.

Finally, next steps for this work are articulated and the overall results of the scoping study are summarized.

Figure ES-1: Sample Representation of Level 2 Approach Spectrum

ACRONYMS

ACL	Accident Class (in US APWR terminology)
ADAPT	Analysis of Dynamic Accident Progression Trees
ADS	Accident Dynamics Simulator
ANS	American Nuclear Society
APB	Accident Progression Bin
APET	Accident Progression Event Tree
AP1000	Westinghouse's Advanced Passive 1000 Design
ASME	American Society of Mechanical Engineers
ASP	Accident Sequence Precursor
ATD	Atmospheric Transport and Dispersion
ATHLET	Analysis of Thermal-Hydraulics for Leaks and Transients
BEIR	Committee on the Biological Effects of Ionizing Radiation
BEMUSE	Best Estimate Methods, Uncertainty and Sensitivity Evaluation
BET	Bridge Event Tree
BNL	Brookhaven National Laboratory
CCI	Core Concrete Interaction
CDF	Core Damage Frequency
CE	Combustion Engineering
CET	Containment Event Tree
CFR	*Code of Federal Regulations*
CORRAL	Containment of Radionuclides Released After LOCA
CPET	Containment Phenomena Event Tree
CSAU	Code Scaling, Applicability, and Uncertainty
CSET	Containment Systems Event Tree
CSNI	Committee on the Safety of Nuclear Installations
DDET	Discrete Dynamic Event Tree
DENDROS	Dynamic Event Network Distributed Risk-Oriented Scheduler
DesktopPA	Desktop Performance Assessment
DET	Decomposition Event Trees; also used in other documents for Dynamic Event Tree
DETAM	Dynamic Event Tree Analysis Method
DYLAM	DYnamic Logical Analytical Methodology
EAL	Emergency Action Level
EI	Exposure Index
EIS	Environmental Impact Statement
EOPs	Emergency Operating Procedures
EP	Emergency Planning
EPGs	Emergency Procedure Guidelines
EPR	Evolutionary Power Reactor
ET	Event Tree
FES	Final Environmental Statement
FY	Fiscal Year
GEIS	Generic Environmental Impact Statement
GRS	Gesellschaft für Anlagen und Reaktorsicherheit
IDAC	Information, Decision, and Actions in a Crew context
INL	Idaho National Laboratory
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination: External Events
IRSN	L'Institut de Radioprotection et de Sûreté Nucléaire
LANL	Los Alamos National Laboratory
LERF	Large Early Release Frequency
LHS	Latin Hypercube Sample
LRF	Large Release Frequency
MAAP	Modular Accident Analysis Program

MACCS	MELCOR Accident Consequence Code System
MARCH	Meltdown Accident Response CHaracteristics
MCCI	Molten Core Concrete Interaction
MCDET	Monte Carlo Dynamic Event Tree
MELCOR	not an acronym
NE&D	Nuclear Engineering & Design
NEA	Nuclear Energy Agency
NPP	Nuclear Power Plant
NRC	US Nuclear Regulatory Commission
PB	Peach Bottom
PDS	Plant Damage State
PDS-ET	Plant Damage State Event Tree
PORV	Pilot (or Power) Operated Relief Valve
PRA	Probabilistic Risk Assessment
RCF	Release Category Frequency
RCS	Reactor Coolant System
SAMAs	Severe Accident Mitigation Alternatives
SAMDAs	Severe Accident Mitigation Design Alternatives
SAMGs	Severe Accident Management Guidelines
SAPHIRE	System Analysis Programs for Hands-on Integrated Reliability Evaluations
SASM	Severe Accident Scaling Methodology
SDP	Significance Determination Process
SECY	Office of the Secretary of the (US NRC) Commission; shorthand for Commission Paper
Seq	Sequoyah
SM2A	Safety Margins Applications and Assessment
SNL	Sandia National Laboratories
SOARCA	State-of-the Art Reactor Consequence Analysis
SPAR	Standardized Plant Analysis Risk models
SSC	Systems, Structures and Components
ST	Source Term
STC-ET	Source Term Category Event Tree
STG	Source Term Group

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1 INTRODUCTION AND BACKGROUND

1.1 Project Genesis

In 2007, the agency issued SECY-07-0192, "Agency Long-Term Research Activities for Fiscal Year 2009" (NRC, 2007). This document identifies a number of scoping studies which are exploratory in nature, to be initiated in Fiscal Year (FY) 2009. One of these studies was entitled, "Advanced Modeling Techniques for Level 2/3 PRA." The write-up associated with this activity cited a number of possible benefits to advancement, and called for a staff-led White Paper and an associated workshop. The current document represents the White Paper called for in SECY-07-0192, and the solicitation of external feedback covered in Section 5 of this document represents the workshop.

> Definition of long-term research from SECY-07-0192:
>
> *"forward-looking regulatory research performed to provide fundamental insights and technical information, or address potential technical issues or identified gaps to support anticipated future (> 5 years) NRC needs."*

1.2 High-Level Objectives/Considerations

This document attempts to scope out Level 2/3 PRA methodologies that might prove fruitful in the long-term (e.g., 5+ years) for advancing the methodology used for Level 2/3 PRA modeling. An advanced methodology should have as many of the following characteristics as possible:

- Reduces reliance on unnecessary modeling simplifications and surrogates (i.e., more phenomenological)
  - E.g., tracks actuation and securing of containment sprays over time and accounts for this in considering phenomena like hydrogen combustion on a sequence-by-sequence basis, rather than only looking at system availability and its effects in a generic fashion
- Addresses methodological shortcomings identified by the State-of-the Art Reactor Consequence Analysis (SOARCA) project
  - E.g., accumulation of conservatisms
  - E.g., lack of consistency in accident progression timings and fission product retention mechanisms with contemporary deterministic calculations
- Improves treatment of human interaction and mitigation
  - E.g., allows the practitioner to explicitly address variability in operator response, including feedback between the accident progression and the implementation of the accident procedures (e.g., Severe Accident Management Guidelines (SAMGs))
- Makes process and results more scrutable
  - E.g., the progression of the accident (including quantitative timing of events) for a given sequence is intrinsic in the model
  - E.g., the criteria and rationale for binning of accident progression end-states in to release categories is more discernable
- Allows for consideration of alternative risk metrics
- Leverages advances in computational capabilities and technology developments, but is computationally tractable
- Allows for ready production of uncertainty characterization
  - E.g., provides the capabilities needed to directly incorporate model and parameter uncertainty

- Permits simplification for regulatory application at a later time (i.e., after it has been sufficiently developed and applied)

A methodology that meets most or all of these goals should be well-suited for evolving the application of Level 2/3 PRA.

Conversely, this document is not an attempt to undermine the existing infrastructure. Any set of tools, methods and guidance has strengths and weaknesses. Logically, any attempt to evolve these tools should attempt to identify and improve upon the weaknesses while retaining as many of the strengths as possible. Lastly, this effort is not an attempt to modify regulatory compliance requirements (e.g., siting criteria). This effort is exploratory in nature, and any recommendations affecting regulatory applications will need to be considered in the context of those applications.

1.3 A Note on Terminology

A traditional PRA is broken in to three major phases. Level 1 PRA covers the period in time from the initiating event to the time of core damage. Level 2 PRA picks up at core damage and follows the accident progression, with its output being the characterization of radioactive releases leaving containment. Level 3 PRA tracks these releases offsite and treats the offsite human health and environmental impacts from the releases, including emergency response (e.g., evacuation). The term Level 2 or Level 3 can be interpreted to mean only that portion of the analysis, as in "…the Level 2 event tree…," or can be used to mean the entire analysis up to that point, as in "…the results of a Level 3 PRA…" Here, they are generally used in the former way to emphasize the particular portion of the calculation being considered, taking for granted the implication that the previous portions have already been performed and that the later portions will be performed. This nomenclature does not restrict the re-visiting of previous portions of the analysis (e.g., Level 1) to address the relevant limitations; it is only used for convenience in specifying the portion of the accident progression being considered.

On a different note, several different terms are used here to describe the individuals who would contribute to a Level 2 and 3 PRA. For the purposes of this document, "severe accident analyst" and "consequence analyst" are terms used to describe individuals who routinely work on deterministic analyses, and may have limited familiarity with the use of deterministic results for PRA. The terms "Level 2 PRA practitioner" and "Level 3 PRA practitioner" are used to describe individuals who routinely take the results of deterministic analyses and translate them in to the PRA models that quantify risk (e.g., event trees), but may not be proficient in the use of the deterministic computer codes and may have limited expertise in the underlying phenomenological behavior. In some instances, a single person or group will fully cover both roles, performing both the deterministic and probabilistic functions. In other instances, these two individuals or groups will be organizationally separated, with the potential for limited interaction to occur. It is concern over this latter situation which prompts the distinction in terms here, and which also plays a role in the original motivation for this work.

2 REVIEW OF PAST USAGE

A number of prominent Level 2 PRAs have been performed for US nuclear power plants over the past three decades. While there are some variations amongst almost every study, it is noteworthy how similar the general framework is for both the Level 2 and Level 3 analyses. Dating back to the 1970s with the WASH-1400 study, the construct of a Level 1 PRA feeding in to an event tree-driven Level 2 PRA feeding in to a set of simulation-based offsite consequence calculations has remained unchanged. Even the second level of detail (e.g., the construct of the event tree, the sampling of weather) has remained relatively similar. Notable differences that do exist include the following:

- NUREG-1150 represented a departure in the amount of detail covered explicitly in the Level 2 event tree itself (i.e., an event tree with ~ 100 top events versus event trees with many fewer top events). Most, if not all, subsequent Level 2 PRAs have reverted back to smaller event trees.

- There is extensive variation in the content of the containment event trees (CETs), and whether a common CET is used for all sequence groups (plant damage states) versus different CETs. Regarding CET content, there is variation between whether the CET explicitly captures system availability, human interactions, and phenomenology versus the capturing of one or more of these via bridge event trees, fault tree linking, etc.

- More recent Level 2 PRAs, most notably those developed for new reactors (e.g., AP1000, EPR) are more complex in the underpinnings of the CET, using fault trees or decomposition event trees (DETs) more extensively. The CETs themselves remain relatively simple.

> Key commonalities amongst past studies:
>
> - All past studies have used an event tree framework for the Level 2 portion.
> - All past studies have elected to use sequence binning/grouping and frequency truncation to capture "all" of the risk profile.
> - While PRA tools (e.g., event trees) and accident progression computer programs (e.g., MELCOR) have both been integral to the past analyses, the methodologies used have opted to utilize them in an "offline" fashion.

Appendix A provides more details regarding the survey of past PRAs. The modeling approaches used in these PRAs are a function of the respective constraints that were placed on the effort, including practitioner or organizational tendencies, resource limitations, intended use of results, computational limitations, modeling limitations, etc. In the current effort, the potential for advancing these approaches is being investigated in light of advances in computational power (e.g., faster hardware) and integrated modeling approaches (e.g., modeling of severe accident phenomenology), and the nature of this effort allows for a more inclusive consideration of the potential uses of the results.

3 REGULATORY PERSPECTIVE

3.1 Level 2 Regulatory Considerations

For operating reactors, consideration of severe accident risk was required for plants licensed after 1980 (see Appendix A, pg. 22), per the Commission's Policy Statement entitled, "Nuclear Power Plant Accident Considerations Under the National Environmental Policy Act of 1969 (NRC, 1980)[1]. In addition, operating reactors developed a Level 2 PRA in response to NRC Generic Letter 88-20 (NRC, 1988) (see Appendix A, pg. 25). In addition, plants seeking license

[1] Additional (subsequent) requirements were developed for high-population sites. See (NRC, 1981) for more information.

renewal have extended their Level 2 models to include offsite consequences (essentially a Level 3 PRA) in order to provide offsite population dose and economic impact information needed for cost-benefit analyses of severe accident mitigation alternatives (see Appendix A, pg. 27). Finally, for other regulatory activities (i.e., risk-informed applications and operating oversight), operating reactor licensees are required to estimate the large early release frequency (LERF), and this is usually accomplished using a simplified, conservative approach.

Several important differences exist for new light-water reactors to be deployed in the coming years. First, all plants will have had a Level 2 PRA performed at the initial licensing stage as part of the design certification / combined license application review (10 CFR 52.47(a)(23) and 10 CFR 52.79(a)(46)). Effectively, this stage also includes a Level 3 PRA of some sort, to support the SAMDA cost/benefit analysis. Second, the Level 2 metric for new reactors is large release frequency (LRF), rather than LERF. Finally, all licensees will be required to develop (and maintain) a plant-specific Level 2 PRA by the time of first fuel load (10 CFR 50.71(h)(1)).

The draft ASME standard for non-light water reactors (e.g., high-temperature gas reactors) replaces the CDF and LERF metrics with a release category frequency (RCF) metric. In essence, this reflects a removal of the interface between classical Level 1 and Level 2 PRA, while promoting a continued interface between Level 2 and Level 3 PRA. Note that this standard is draft, and subject to change.

Lastly, the agency sometimes employs Level 2/3 PRA in its internal decision-making. Two notable examples are cost/benefit analyses associated with plant modification requirements and resolution of generic safety issues.

3.2 Use of Level 2 in SPAR / SDP

The NRC maintains a set of plant-specific PRA models (a.k.a., Standardized Plant Analysis Risk (SPAR) models) which are used by NRC staff in support of risk-informed activities related to the inspection program (i.e., the Significance Determination Process (SDP)), the incident investigation program, license amendment reviews, performance indicator verification, the Accident Sequence Precursor (ASP) program, generic safety issues, and special studies. These models also support evaluations of operating experience, thereby providing the agency with the ability to analyze operating experience independently of licensees' risk assessments.

There have been a number of attempts to bring Level 2 modeling in to the agency's SPAR models. Originally, Level 2 models were developed for several plants which mimicked the NUREG-1150 models. Later, simplified models were developed in reaction to criticisms that the earlier models were too complicated for practical use. However, these models still suffered from the complexity in maintaining a separate Level 2 model that received far less maintenance attention and application use than the corresponding Level 1 model, but was still affected by changes to the Level 1 model. In an attempt to address this, integrated models were developed for three plants, as documented further in Appendix B (see pg. 32). Figure 1 captures some of this history, while Figure 2 presents an envisioned path forward as of Fall 2008. Others have also proposed structures similar to that in Figure 2 (see Torri, 2009).

Currently, the SDP notebooks use a multiplication factor on a core damage frequency sequence basis to estimate the increase in LERF for Phase 2 SDP evaluations. LERF multiplication factors for initiating events or plant conditions are provided in Inspection Manual Chapter 609 Appendix H and in NUREG-1765, “Basis Document for Large Early Release Frequency (LERF) Significance Determination Process (SDP)” (NRC 2002a). The SDP notebooks also incorporate factors for high and low pressure sequences. There is currently an effort underway to apply the SDP notebook LERF multiplication factors to Level 1 SPAR models in the SAPHIRE Version 8 SDP interface. The results for analysis using these factors will be characterized as “Screening LERF” to distinguish them from results which would be obtained from a LERF SPAR model analysis.

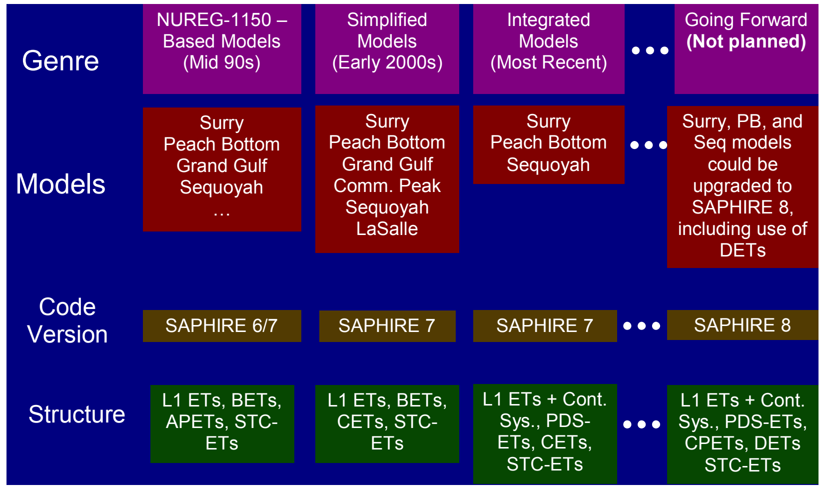

Figure 1: History of Level 2 SPAR

3.3 ANS Level 2 and Level 3 Standards

The current combined ASME/ANS PRA standard covers Level 1 and a partial Level 2 (LERF) approach. For new LWRs, LERF has been replaced by LRF and the requirement is a full, rather than partial, Level 2. For this reason, a separate Level 2 (and a separate Level 3) standard are under development by ANS. Like the existing standard, the current drafts of the standards define the requirements for three levels of “capability category,” allowing for different levels of rigor based on the application. Also like the existing standard, the new standards assume the framework of a traditional approach (e.g., a static containment event tree methodology for Level 2), but do not preclude other approaches. The new standards are draft, and the attributes defined above are subject to change as requirements in the standard are refined, and eventually subjected to independent peer review.

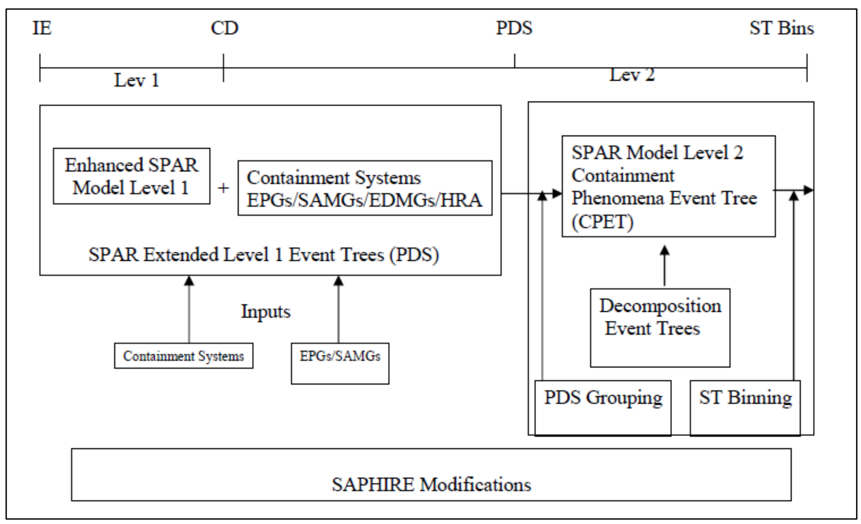

Figure 2: Potential SPAR Level 2 Approach (from Sherry, 2008)

4 RECENT DOMESTIC AND INTERNATIONAL DEVELOPMENTS

A number of recent development efforts have also been investigated. Some of these efforts use the traditional PRA framework as a basis, and make relatively small changes to particular aspects of it. These include the use of extended Level 1 models, decomposition event trees, Bayesian networks[2] in lieu of fault trees, and event tree development based on functional binning of accident signatures. Summarized information regarding use of the traditional Level 2 methods is also available in a 2007 Nuclear Energy Agency report on recent development in Level 2 PRA, including information on Level 1/2 PRA interface issues, modeling of human intervention, and Level 2/3 PRA interface issues (NEA, 2007).

Other approaches investigate simulation-based methods, using system codes[3] in conjunction with dynamic event trees. Dynamic event tree approaches typically have three fundamental characteristics in terms of user-specific control:

- Branching and stopping rules, to guide the initiation of branch points and the cessation of very low probability or no-release sequences;
- A plant simulator (i.e., an accident analysis computer code which may or may not include a crew simulator); and
- Probability assignment rules to dictate the assignment of branch-point probabilities.

The way in which each of these three characteristics is handled defines the differences between the approaches.

In addition to these distinctions, the motivation for tool development will have a fundamental effect on the tool's scope and strengths. For instance, the dynamic event tree approach being developed by the Ohio State University (ADAPT/MELCOR) is an approach designed with Level 2 applications in mind, while the dynamic event tree approach being developed by the University of Maryland (ADS-IDAC) is designed more for exploration of Level 1 PRA operator action effects.

An approach that seems to have received less development effort is the use of sampling-based simulation without the use of an event tree. This approach is akin to the way that one might address uncertainty quantification for a deterministic calculation (and in fact, the use of probabilistic tools for accomplishing uncertainty quantification has been a point of interest since at least the late 1980s). For use in Level 2, the key difference would be the need to also account for variability in system availability and human action. The application to Level 2 would also be broader in the sense that past use of these types of methods has generally focused on the assessment of a particular scenario or a particular phenomena.

Each of the above approaches represents a distinct point within a spectrum of possible approaches that can be taken, and this notion is key to the discussion in later sections of this document. Appendix B provides more details regarding each of the approaches/activities that were investigated.

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2 Bayesian Network (or Bayes Net for short) refers to a specific type of modeling approach that relates various inter-connected elements of a system or process by the probabilities or degree of belief (e.g., subjective probabilities) that relate the different constituent elements. Their strength lies in their ability to relate complex multivariate dependencies.

3 The term system code is used in this document to refer to a computer program that solves the governing equations for fluid flow and heat transfer to predict the system-level behavior of a nuclear power plant. Typically a severe accident / Level 2 PRA tool would also model structural response, chemical reactions (e.g., cladding oxidation, hydrogen generation), fuel degradation and relocation, fission product volatilization and deposition, etc. For these applications, the NRC develops and maintains the MELCOR code.

5 EXTERNAL STAKEHOLDER FEEDBACK

On March 9th, 2009, a number of invited experts representing a broad range of interests gathered in person, or via phone, to discuss issues related to modeling in Level 2 and Level 3 PRA. The purpose of the meeting was to gather input for this particular activity, but the materials provided and the meeting moderation were intentionally focused on gaining broad feedback. It was not the intent to reach consensus on any specific issues as part of this brainstorming session. Even so, there are certain issues where some level of agreement was reached and these are provided in the following bullets:

- Binning effects at both the front-end and back-end of the Level 2 analysis can be reduced by the treatment of more PDS (when used) and more release categories.
- The treatment of operator response (including HRA) and accident management in Level 2/3 PRA would benefit from additional modeling attention. The tracking of EOPs/SAMGs synchronously with accident progression is beneficial.
- The treatment of parametric and modeling uncertainty can be improved, and in the long-term would benefit practitioners in terms of focusing attention on uncertain risk-significant aspects of the analysis.
- The view of uncertainty fundamentally affects the preferred construction of a Level 2 methodology.
- Given the desire to use Level 2/3 results for an application other than regulatory risk metric compliance, the fidelity of existing models could be improved.
- Resource limitations will always necessitate choices between completeness and fidelity. These limitations should be recognized and research should aim to improve guidance on where simplification is acceptable versus where it is not.
- The running of physical models (e.g., MELCOR) continues to be an important aspect, in terms of capturing non-intuitive effects and ensuring that the Level 2 PRA does not become too out-of-step with severe accident research
- Specific Level 3 modeling issues could be informed by more attention (e.g., non-nuclear seismic consequences, alternative risk metrics).
- The desired end-use of the results should fundamentally affect the intermediate modeling decisions.

Other points that were raised where a partial consensus was not reached include:

- The view of phenomenological and structural response uncertainty (and their associated treatment in split fractions) as either primarily stochastic or epistemic.
- The specific approach that should be taken in improving Level 2/3 PRA model fidelity.
- The degree to which upgrades in most Level 3 aspects (e.g., atmospheric transport and dispersion (ATD) modeling) would affect the end results of interest.
- The view of how much reliance should be placed on the practitioner versus a computer simulation.

A more expansive cataloguing of ideas discussed during this meeting is provided in Appendix C.

6 OVERVIEW OF DISCRETE APPROACH CATEGORIES

As noted previously, there is a broad spectrum of approaches that can be taken. For the purposes of this paper, that spectrum will be viewed as one which ranges from the traditional methodology used by most of the studies described in Section 2 and Appendix A to a purely simulation-based approach that analyzes an exhaustive set of scenarios based on probabilistic treatment of each system, operator, and phenomenological interaction.

As an illustration of this concept for the Level 2 portion of the analysis, Figure 3 presents a cartoon of this idea. Four broad approaches are selected for further consideration, all of which are based in one way or another on previously suggested methods. In reality, each approach is a mixture of numerous related approaches, and even after a particular approach has been selected for evaluation, many additional questions will remain regarding the second-level details for the selected approach. For ease in discussion here, the four broad approaches are assigned the following nomenclature:

1 Modified traditional approach – This is a CET-based approach which could utilize fault trees, decomposition event trees, and/or Bayes Nets as the basis for top events.
2 Hybrid event tree approach – This is an event tree-based approach in which multiple event trees are constructed based on specifics of the accident signature, and the system code analysis results are used iteratively to adjust the event trees.
3 Dynamic event tree simulation approach – This is an approach whereby an executive program uses branching rules to develop a time-based event tree using the real-time results of a system code and crew module.
4 Sampling-based simulation approach – This is an approach where the system code input parameters, phenomenological models, and operator action logic have probability distribution functions that are randomly sampled, for each of a large number of calculations, to arrive at a distribution of results.

Figure 3: Sample Representation of the Level 2 Approach Spectrum

Figure 4 shows an alternate representation (as compared to Figure 3) for how the approaches represent evolution. The important aspect of this cartoon is its attempt to present the approaches in a manner whereby one approach can logically lead to the next. In this view, certain variations of one approach class are nearly identical to certain variations of another approach class. For example, one could select a variation of Approach 2 that is nearly identical to a variation of Approach 3, except that construction of the event tree would be handled manually rather than automatically. Similarly, one could identify a variation of Approach 3 that is nearly identical to a variation of Approach 4, except for the use of event trees to track sequence probabilities and treat branching.

All four proposed approaches would use the same general framework for the Level 3 portion of the calculation as is currently used. However, issues such as effects of external events (e.g., earthquakes, external floods) on emergency response, propagation of uncertainties through the Level 2/3 interface, atmospheric dispersion modeling, economic modeling and/or other methods and data issues would warrant re-visitng. A literature review of the use of PRA tools in post-event analysis, regulation, and guidance development (including preparedness, response, recovery, and reconstitution) is provided in (Siu, 2006).

The following pages articulate the major strengths and weaknesses of the various approaches being considered. In addition, they provide (via text boxes) the basic steps envisioned for a representative version of the approach. Last, for each approach a representative version of the approach is shown graphically to illustrate the idea. It is important to keep in mind that these devices are intended to help the reader understand the general Level 2 framework that is being proposed, not to present a definitive treatment.

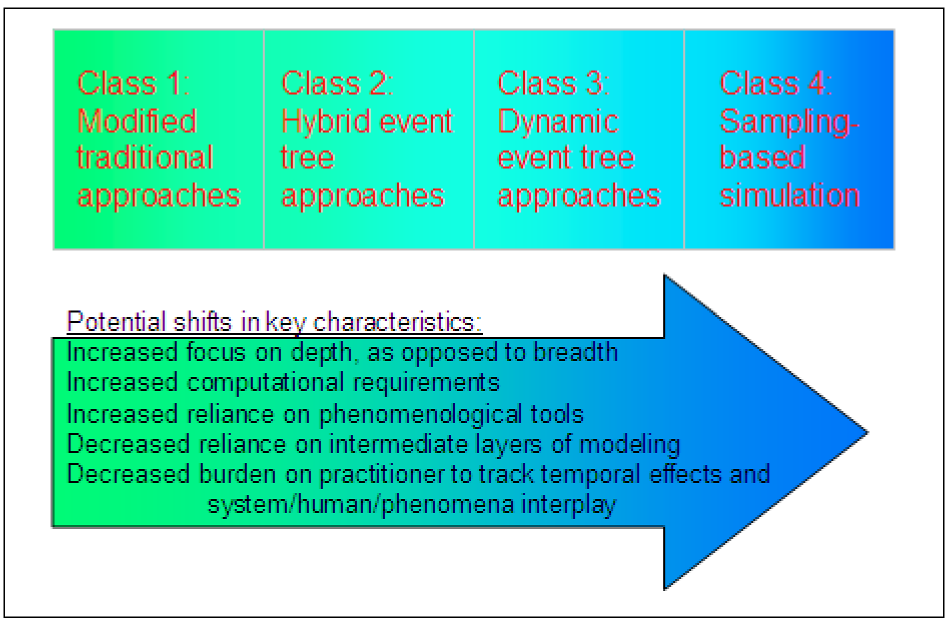

Figure 4: Alternate Representation of the Level 2 Approach Spectrum

Terminology Tip: Modified traditional approach – This is a CET-based approach which could utilize fault trees, decomposition event trees, and/or Bayes Nets as the basis for top events.

Approach 1: Modified Traditional Approach

- Traditional approaches are by far the most widely-accepted framework for conducting a Level 2/3 PRA analysis, and allow for book-keeping of a large number (thousands) of sequences covering the full range of initiating events. By virtue of its similarity to these approaches, the modified traditional approach would retain these characteristics.
- This approach relies heavily on the expertise of the Level 2/3 PRA or accident progression/consequence analyst who is constructing the event trees and performing the binning (including the supporting logic rules) to account for numerous (often non-linear and dynamic) effects in operator action and accident phenomenology.
- This approach leverages technology advances to some degree (Bayes Nets) but not computational advances.

Sample Steps for Approach 1:

- Use an Extended Level 1 SPAR model to perform Level 1 analysis
- Group extended Level 1 end states as necessary
- Develop phenomenological CET with supporting fault trees, decomposition event trees, or Bayes Nets for each top event
- Use event tree-level quantification
- Bin resulting release sequences in to 10s of release categories
- Run MELCOR calculations to support above binning and event tree construction, and to develop release fractions and timings for input to Level 3
- Run MACCS2 calculation for each release category + needed variations
- Use LHS for SPAR and small-sample LHS (e.g., 100s of samples) for MELCOR/MACCS to estimate uncertainty

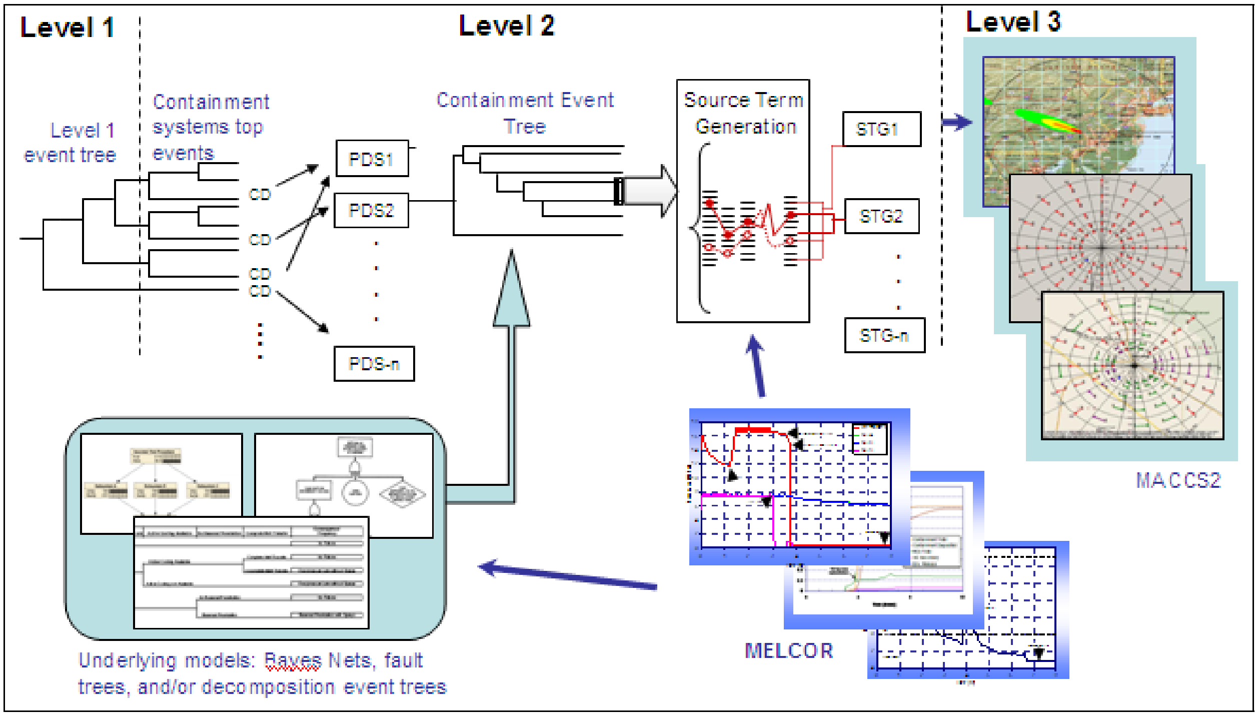

Figure 5: Sample for Approach 1

Terminology Tip: Hybrid event tree approach - This is an event tree-based approach in which multiple event trees are constructed based on specifics of the accident signature, and the system code analysis results are used iteratively to adjust the event trees.

Approach 2: Hybrid Event Tree Approach

- This approach retains many of the features of the widely-accepted traditional approach, and can be used to cover a large number of sequences
- This approach leverages technology advances, as well as computational advances
- This approach increases the incorporation of phenomenological modeling in to the event tree structure. It may or may not improve scutability, depending on how branching is handled.
- The approach promotes a more common view of accident progression between the Level 2/3 PRA practitioner and the severe accident/consequence analyst.

Sample Steps for Approach 2:

- Utilize plant-specific SPAR model results to identify all key sequences
- Run MELCOR calculations to establish Level 1 outcome and quantify timings (including EALs) for key sequences
- Group dominant core damage sequences in to plant damage states based on system availability (10s of groups)
- Run a base MELCOR calculation for each plant damage state
- Select important points in the accident progression (including key operator interactions) and run additional MELCOR calculations based on these variations
- Construct event tree for each PDS that captures these different sequences, including the derivation of split fractions, which could be based on fault trees, DETs or Bayes Nets (10s of ETs)
- Use MELCOR results to bin ET end-states based on source term similarity (10s of release signatures)
- Run MACCS2 calculation for each release signature bin + needed variations
- Use LHS for SPAR and small-sample LHS (e.g., 100s of samples) for MELCOR/MACCS to estimate uncertainty

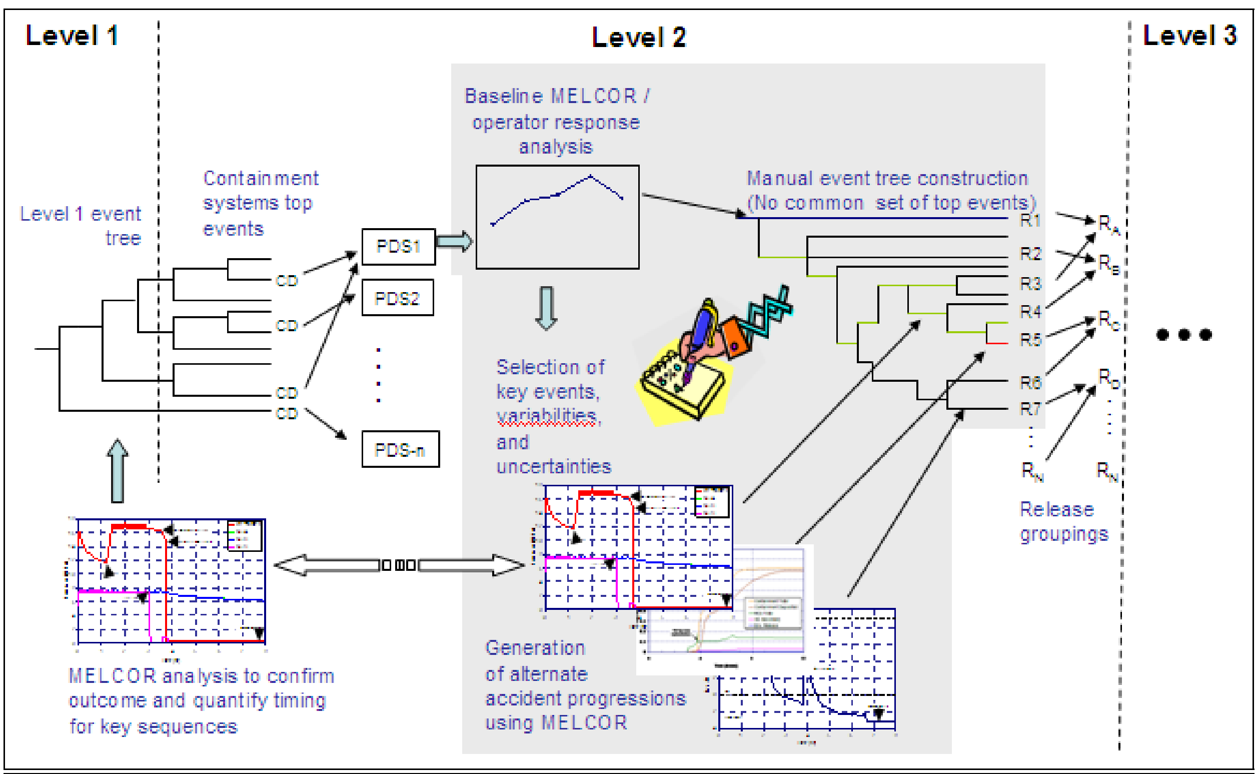

Figure 6: Sample for Approach 2

Terminology Tip: Dynamic event tree simulation approach - This is an approach whereby an executive program uses branching rules to develop a time-based event tree using the real-time results of a system code and crew module.

Approach 3: Dynamic Event Tree Approach

- This approach would be less familiar to the traditional Level 2 PRA practitioner, but readily understandable.
- This approach leverages both technology and computational advances.
- This approach is likely not computationally tractable for covering the entire risk profile.
- This approach would allow for incorporation of models that treat operator response dynamically, recognizing that these models would require significant development and monitoring.
- This approach would be more phenomenological, in that the mechanistic severe accident analysis would be directly generating event tree branching events (albeit based on practitioner-specified rules).
- This approach might or might not be more scrutable, as compared to currently employed techniques, depending on developments in presenting results.

Sample Steps for Approach 3:

- Develop/employ a driver routine for handling branching rules, stopping rules, and input/output processing
- Run MELCOR calculations to establish Level 1 outcome and quantify timings for key sequences
- Bin Level 1 end-states based on system availability and expected accident progression behavior
- Run a baseline MELCOR calculation, and use as a guide in identifying key events / uncertainties; develop rule sets for driver routine
- Run driver routine in combination with MELCOR and the crew module to automatically generate dynamic event trees for each PDS (100s of end-states per PDS)
- Perform grouping of release signatures (10s of release signatures)
- Run MACCS2 calculation for each release signature bin + needed variations
- Use LHS for SPAR and small-sample LHS (e.g., 100s of samples) for MELCOR/MACCS to estimate uncertainty

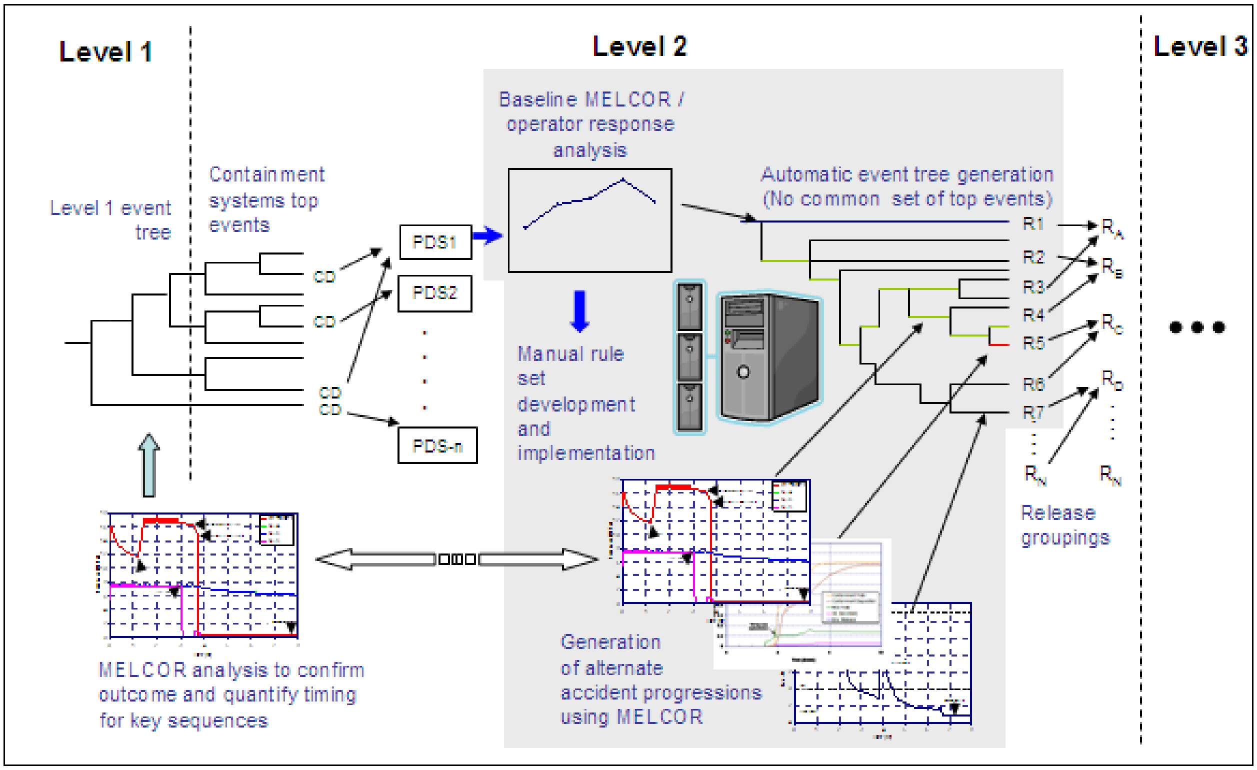

Figure 7: Sample of Approach 3

Terminology Tip: Sampling-based simulation – This is an approach where the system code input parameters, phenomenological models, and operator action logic have probability distribution functions that are randomly sampled, for each of a large number of calculations, to arrive at a distribution of results

Approach 4: Sampling-Based Simulation Approach

- This approach would be unfamiliar to the traditional Level 2 PRA practitioner, but very familiar to an accident progression analyst.
- This approach leverages both technology and computational advances.
- This approach is likely to not be computationally tractable for covering the entire risk profile. Smart-sampling techniques will need to be developed to ensure that combinations of initial and boundary conditions are physically reasonable.
- Like Approach 3, this approach allows for incorporation of models that treat operator response dynamically, recognizing that these models require significant development and monitoring.
- This approach would be much more phenomenological, in that all sequences would be directly solved by the severe accident code.
- This approach might or might not be more scrutable, as compared to currently employed techniques, depending on developments in presenting results.

Sample Steps for Approach 4:

- Utilize plant-specific SPAR model results to identify the key sequences of interest
- Run MELCOR calculations to establish Level 1 outcome and quantify timings for key sequences
- Bin Level 1 end-states using system availability and expected accident progression behavior
- For a given PDS, sample MELCOR inputs (initial and boundary conditions) using LHS
- Execute the calculation, sampling modeling and operator uncertainties during execution
- Repeat numerous times (e.g., 100s of times per PDS), until results show convergence (i.e., reproducibility)
- Repeat for other PDSs
- Bin end-states based on source term similarity
- Run MACCS2 calculation for each release signature bin + needed variations
- Use LHS for SPAR and small-sample LHS (e.g., 100s of samples) for MELCOR/MACCS to estimate uncertainty

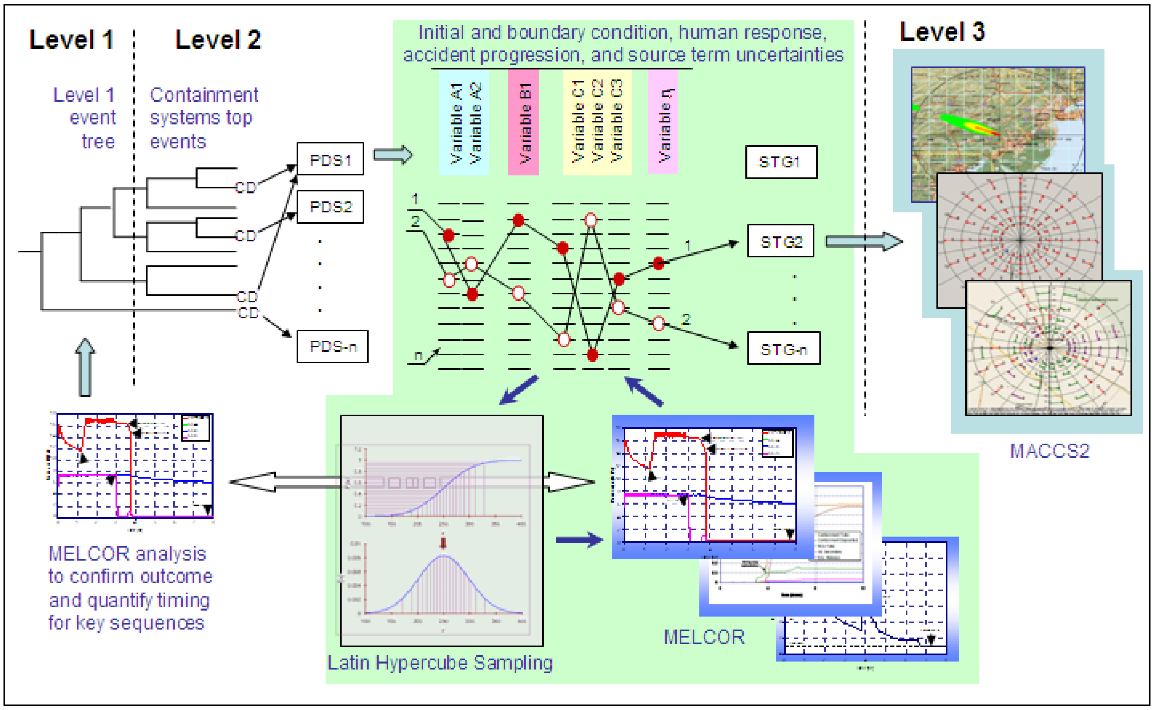

Figure 8: Sample of Approach 4

Digesting the above descriptions can be difficult, because (a) the characteristics are loosely defined and (b) the similarities and differences between the methods aren't directly compared. For this reason, Table 1 attempts to directly compare these characteristics. Even so, the characteristics for a given approach are subjective, as nearly any approach can be modified in a given aspect to better conform to a different approach.

Table 1: Comparison of Different Approaches

Characteristic	Approach 1	Approach 2	Approach 3	Approach 4
Starting point	Level 1 PRA results			
PDS Grouping	Bridge trees, transfer trees, or extended level 1 trees .OR. specific Level 1 end-states for a limited study			
Confirmation of Level 1 end-states	Practitioner experience	Distinct MELCOR analysis		
Consideration of Level 1 variations	Level 1 PRA	Level 1 PRA .OR. limited consideration in MELCOR model		
Event tree structure	Traditional	Static, based on MELCOR results	Dynamic	N/A
Determining outcome for specific events	Logic model	Based on computer simulation		
Operator response	Practitioner-added top/basic events (current HRA best practice)		Limited rule sets or cognitive crew model	
Probabilistic / deterministic coupling	Decoupled	Manually coupled	Dynamically coupled	Intrinsically coupled
Source term grouping	Source term event trees / release category binning			
Consequence calculations	MACCS2			
Long-term recovery and offsite assets	Probabilistic weighting in Level 2			
Evacuation modeling	MACCS2			
Risk integration	Traditional propagation			Even weighting of sample results

Finally, a simplistic graphic of the Level 3 step is offered in Figure 9, since this step is simplified or omitted in Figure 5 through Figure 8.

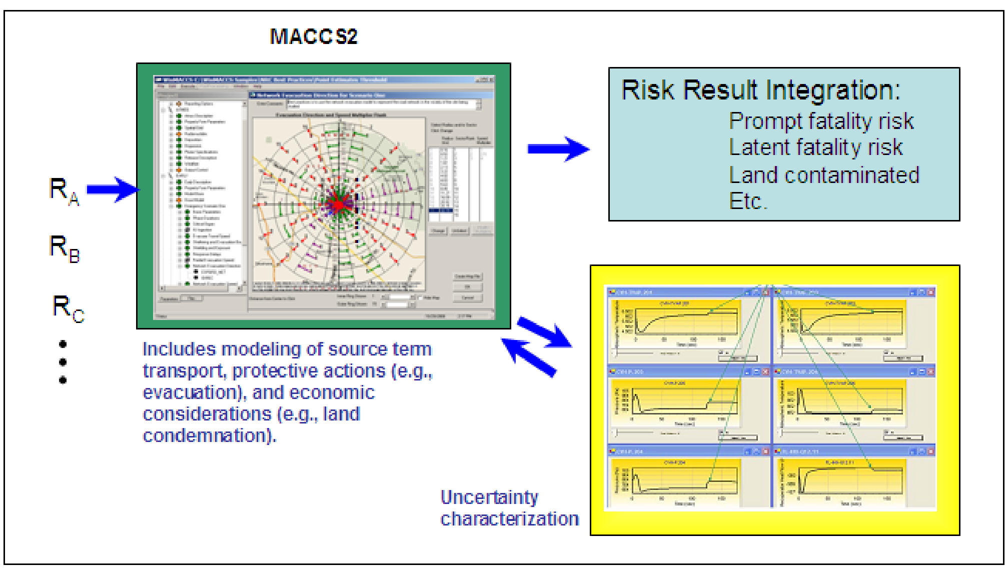

Figure 9: Sample of the Level 3 Framework

7 SELECTION OF A CANDIDATE APPROACH

In selecting an approach, it is first useful to articulate what needs we are trying to fill, versus what needs are secondary, as certain approaches will lend themselves better to certain needs. Questions that should be considered include:

1. What are the fundamental characteristics being sought?
2. What does the alternative method offer over the traditional approach, and why is this enhancement needed/beneficial?
3. Is the intent to produce a methodology which will address the entire risk profile, or to produce a methodology which allows one to investigate particular scenarios to a higher fidelity (e.g., a particular set of scenarios relevant for a generic issue)?
4. What is the starting point: a blank sheet or an already completed Level 2/3 PRA based on a traditional approach?
5. What additional work is needed to fully develop and implement a given approach, and what intermediate benefits will be gained along the way?

For the first question, a list of attributes is provided in Section 1.2. The second question is closely tied to the first, but is more application-specific. For some applications, the existing approach may be suitable (e.g., the cost to refine does not justify the gain in accuracy, the decision is not affected by the model's limitations), while in other applications the identified shortcomings may not be justifiable. In general, this effort is focused on research use, where the desire for realism is greater. All of the characteristics identified in Section 1.2 are areas in which the existing methods are viewed to have shortcomings, and in the context of a research application, these are the areas which we will try to inform.

With regard to the third and fourth question above, operating reactors all maintain a living Level 1 PRA, with some form of Level 2 capability (of widely varying rigor). For operating reactors that have gone through license renewal, a more extensive Level 2 PRA may exist, but perhaps not be maintained. For new reactors, a generic Level 2 will have been performed as part of design certification, but an as-built, plant-specific Level 2 PRA will not exist until sometime before the first fuel load. For these reasons, it will be assumed that a Level 2 of reasonable pedigree will be the starting point, but that the Level 2 may not have been updated within the past 5 years. As such, the idea will be to develop a methodology that can be used to refine the portions of the model (e.g., specific plant damage states) that are viewed as more risk-significant or are of interest for a particular application, but not to necessarily develop a methodology that is capable of addressing the entire spectrum of Level 1 end-states feeding in to the Level 2. It is also important to consider that the existing Level 1 PRA (or SPAR model, in the case of NRC) is the product of the constraints placed upon its developments. As such, it may have simplifications or conservatisms that are important to the inputs for the Level 2 PRA. Examples include the use of a prescribed, conservative timing for specific consequential failures (e.g., reactor coolant pump seal failure) and the lack of partial crediting for operator or mitigative actions that may alter the time to core damage or vessel breach. In conducting the Level 2 PRA, it is important to understand these simplifications or conservatisms, such that they can be either (a) accounted for in developing the Level 2 PRA and/or (b) modified such that the inputs to the Level 2 PRA are more rigorous. This issue is discussed further in Secition 8, "Variations of the Candidate Approach."

The fifth question has been implicitly considered in developing the spectrum of approaches described in Section 6. In general, the evolution of methods depicted in Figure 3 and Figure 4 are also believed to roughly correlate to the amount of effort needed to fully develop and implement the approaches. Regarding the potential benefits that can be realized along the way, this issue is addressed briefly in Section 9 ("Next Steps") for the selected approach.

The four approaches described in Section 6 were qualitatively assessed against the desirable attributes identified in Section 1.2, including the ranking of the various approaches for each characteristic. Based on this assessment, Approaches 3 and 4 are the best suited for addressing the issues that have been presented. Both approaches leverage computational and technology advances. Specifically, they both improve the treatment of event progression by explicitly considering temporal evolution and directly employing accident simulation tools. They also provide greater flexibility in treating operator action by dynamically coupling the operator response, system response, and event progression. Finally, both approaches may result in greater scrutability compared to currently used methods, if appropriate techniques for presentation of results are developed.

As a tactical decision, Approach 3 is recommended for the focus of future work, for the following reasons:

1. Approach 3 provides more opportunity for leveraging work by others in the PRA community. Specifically, a larger body of work exists in treating accident sequences within a probabilistic framework using dynamic event trees (including the consideration and treatment of rare events and dependencies). These efforts also provide more operating experience in addressing difficulties that arise when coupling the various pieces (e.g., the coupling of an operator model to a system code).
2. Approach 3 is judged to be superior from a computational tractability stand-point. The use of an event tree structure prevents continuous repetition of portions of the calculations that are substantively the same[4]. This allows for greater reproducibility of the results (e.g., release timing) for a fixed number of simulations. Similarly, Approach 3 increases the practitioner's involvement (by virtue of branching rule development) from the aspect of ensuring that the sampling of different distributions does not result in non-physical boundary conditions (in the absence of correlation coefficients).
3. Approach 4 is most like the type of analysis that will be performed as part of the uncertainty characterization phase of the SOARCA project (even though that effort is aimed at deterministic analysis), and operating experience on the strengths and weaknesses will be gained from that endeavor. Those lessons learned will benefit any future work done under this project as well. As such, that effort will provide the forum for partially scoping out that approach.

For the above reasons, Approach 3 is the one recommended for future work, with recognition of the fact that Approach 2 (as a stepping-stone) and Approach 4 (as a potential later evolution) also have merit.

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[4] This refers to the fact that Approach 3 follows a sequence and creates branch sequences where necessary, while Approach 4 is designed to repeat the entire calculation hundreds of times, each time with a new set of initial and boundary conditions. The development of smart-sampling techniques might address this shortcoming.

8 VARIATIONS OF THE CANDIDATE APPROACH

Once a candidate class of approaches has been selected, it is important to investigate the variations possible in terms of sequence binning, operator response modeling, top event selection / branching decisions, quantification, uncertainty propagation, Level 2/3 interface issues, emergency preparedness implementation, differences for addressing external events, presentation of results, etc. The majority of this will be deferred to the (proposed) subsequent effort, but some thoughts are offered below.

One step in this process, which would be performed as part of the subsequent contractor-led work, would be to better understand the relevant efforts that have already been undertaken elsewhere. Appendix B provides some information about some of these efforts, but it only addresses them at a cursory level to support the cross-class comparison performed above. A more detailed investigation of the relevant approaches is necessary, and a more rigorous consideration of their relative pros and cons. Presumably the descriptors used could include things like:

- What problem(s) are the developers trying to address?
- What phenomena/phenomenological issues are they treating, and what other problems could they potentially treat within their framework?
- What issues are they not treating?
- What is their computational approach, and their state of implementation?
- Regarding implementation, what level of review have the method(s) and model(s) received?

In particular, there might be examples where methods or software packages can be directly or indirectly used. In cases where existing software packages (other than MELCOR and MACCS2 which are fully within NRC's control) exist, the viability of leveraging these tools should be investigated.

In invoking Approach 3, there are many ways to tackle each aspect of the analysis. In the area of modeling human interactions, a particular interest is the ability to accurately model the various paths that result from the operators' execution of the EOPs, SAMGs and security-related mitigation procedures. Significant variations can occur based on the time that actions are taken and any deviations from the procedures (either intentionally or unintentionally). Work at the University of Maryland in this area is briefly discussed in Appendix B. An example of recent work using traditional methods can be found in (Lutz, 2008). As an example of how this issue might be tackled, one might start by developing simple rules by which the operators take (or don't take) specific key actions simply using stylized factors for the probability they will take the action and a stylized distribution for when it will be taken. Subsequent steps could incorporate more actions, performance shaping functions based on accident response, and eventually cognitive treatments.

The issue of what starting point to use may be more application-specific. For a study with a narrow scope, one might be able to directly use core damage end-states as the starting point. For broader-scope studies, some binning will likely be necessary. The extent to which this binning mirrors traditional approaches versus the extent to which it can utilize the nature of the refined event tree treatment to better characterize variability is an issue that needs to be investigated. Similar considerations exist for the grouping of releases for propagation to the Level 3 analysis. One important insight from the external stakeholder meeting is most participants felt that less binning is necessary (i.e., a larger number of bins can be accommodated) on both the front and back-end of the Level 2 analysis, due to increased computational capabilities.

Regardless of whether binning is used at the start of the Level 2 PRA, it is important to ensure that the Level 1 treatment feeding in to the Level 2 analysis is accurate to the resolution required for the application. Simplifications are often made in the Level 1 PRA with regard to success criteria or timing to account for the very large number of sequences that need to be considered. This can lead to conservative identification of core damage sequences and/or conservative views for the timing of core damage or vessel breach used in subsequent plant damage state binning. Since the Level 2 PRA considers a subset of the overall Level 1 sequences, there is an opportunity to refine some of these assumptions and make the Level 2 inputs more realistic. The extent to which this is realized in traditional analyses varies, depending on the application and the developers. Refinement of the Level 2 inputs can be accomplished in a number of different ways (e.g., accounting for timing variability in the determination of CET split fractions), and the description of the various approaches in Section 6 includes this step in an ambiguous manner.

The use of MELCOR in Approaches 2-4 in a manner that directly links a given accident simulation with its corresponding Level 2 sequence provides an opportunity to naturally re-visit the Level 1 portion of the sequence. It is expected that the MELCOR analyses will necessarily start at the initiating event (i.e., “t=0”). As such, it will progress through the portion of the accident preceding core damage (i.e., the Level 1 portion), and provide quantitative timings of key Level 1 events and confirmation of the Level 1 assessment of core damage. However, the boundary and initial conditions for the simulation will generally be those associated with the plant damage state being analyzed, not necessarily with the initial and boundary conditions of a specific Level 1 core damage sequence of interest. To the extent that the initial and boundary conditions associated with the PDS represent well the key Level 1 sequences, the confirmation and quantification of key Level 1 sequences will be achieved. However, it is recognized that the details of particular Level 1 sequences are often lost (i.e., the inherent variability within each PDS is averaged out) when performing the plant damage state grouping. In this case, there are other ways that one can ensure that the Level 2 inputs are consistent and accurate for key sequences. Two methods are described below.

Prior to initiating the Level 2 PRA development, one could systematically re-visit the Level 1 PRA for known issues (e.g., reactor coolant pump seal LOCA) and important sequences (e.g., short-term station blackout). In cases where the Level 1 analysis has made simplifying assumptions, these could be addressed by refining the Level 1 model (e.g., adding a new top event) and/or running additional thermal-hydraulic success criteria analysis to confirm key timings. The advantage of using a tool such as MELCOR for this purpose is that it also allows for post-core damage analysis, such that subsequent timings (e.g., vessel breach) can also be quantified for consideration in subsequent plant damage state grouping.

A second alternative would be to complete the Level 2/3 analysis as previously described for Approach 3, and to subsequently investigate to greater detail the risk-dominant sequences. This requires identification of the most important Level 3 contributors, tracing these contributors back through the Level 2 PRA to identify their Level 1 origins, and then re-quantifying the Level 1 sequences more rigorously to ensure that they were accurately treated by the Level 1/2 interface.

Another issue of interest is how the results of intermediate analysis steps and the final risk results are presented. As pointed out in Section 6, the scrutability of Approaches 3 and 4 will be determined by the development of techniques for presenting the results. Due to their novelty relative to traditional approaches, the techniques currently used to present results (e.g., graphical event trees) may not prove practical. The approaches are expected to provide greater scrutability in the linkage between sequence progression and phenomenological / operator response modeling, but it is incumbent upon the developers to ensure that the appropriate techniques are developed to realize this greater scrutability. Furthermore, the advanced methods may provide the higher fidelity needed to justify the assessment of results using more rigorous risk metrics (e.g., a finer parsing of releases in terms of their offsite effects).

Similar arguments/plans need to be developed for each aspect of the analysis. In doing so, it is useful to keep in mind the notion that methodological investigations should seek to generate practical advice, whenever possible. Consequently, whenever possible, as this (proposed) effort seeks to refine the modeling approach, it should also seek to (a) quantify the effect that incremental method changes are having on the end results, and (b) quantify the change in the relative significance of particular issues (e.g., operator-initiated depressurization during a station blackout). This can be most readily achieved by selecting a particular scenario to be evaluated (e.g., station blackout) as the reference scenario and establishing a baseline using a recently completed traditional Level 2/3 PRA.

9 NEXT STEPS

Based on the discussion provided in this document, work is being proposed to further explore, and develop as practical, Approach 3. Specifically, a contract is recommended to achieve the following:

1) Review and critique this document, from the standpoint of (a) improving the document's utility, including the addition of examples to better illustrate the issues being raised and (b) augmenting (as necessary) the items and issues that should be considered in further developing Approach 3.
2) Develop an expanded summary of the relevant methods that have been developed elsewhere (e.g., ADAPT/MELCOR, ADS-IDAC, MCDET) and generate recommendations for how this work can reasonably leverage that work. Similarly, relevant developments in advanced methods (including those pertinent to Approach 4), and the state-of-the-practice in Level 2/3 PRA should be tracked and considered.
3) Investigate, document, and generate recommendations regarding the variations that should be pursued with regard to the types of issues discussed in Section 8 (e.g., confirmation of Level 1 outcomes/timings for important sequences).
4) Develop, and submit for approval, a plan that lays out the following:
    a) The identification of key individuals (e.g., contractor staff, NRC staff, and/or independent experts) who should review intermediate products to ensure that significant issues are not being neglected or mis-characterized.
    b) A description of how the proposed work leverages and/or relates to other recent developments (e.g., ADAPT/MELCOR).
    c) The establishment of a baseline, including selection of a particular scenario (e.g., station blackout), such that the effects of methods modifications can be quantified.
    d) The plan for what specific attributes the end-product should have (e.g., the platform and message-passing specifications for the executive program).

e) The intermediate steps that will be taken to ensure that practical considerations are being addressed (e.g., conduct of a manual dynamic event tree development exercise akin to Approach 2).
f) The timeframe for developing the necessary capabilities (software or otherwise) for a demonstration tool.
g) The timeframe for the application of that tool to analyze the scenario selected above.
h) The identification of attributes that can be incorporated in to the method to support flexibility (e.g., allow for more manual involvement akin to Approach 2 or more direct sampling akin to Approach 4).

5) The execution of the above plan, including the necessary review.
6) The identification of insights gained from the work, including quantitative effects of the modified method.

If pursued, the contractor undertaking the study should have most or all of the following characteristics:

- Be expert in the analysis of severe accidents at nuclear power plants,
- Be familiar with historical Level 1, Level 2, and Level 3 PRA studies,
- Be knowledgeable with regard to the treatment of severe accident phenomena, human interaction (including emergency response), structural response, and consequence analysis issues in contemporary Level 2 and Level 3 PRAs, and in contemporary deterministic analyses,
- Possess a good understanding of how emergency procedures and accident management guidelines are utilized,
- Be proficient in the analysis of severe accidents using MELCOR, and consequence analysis using MACCS2,
- Possess a good understanding of the human reliability analysis technology, as applied in Level 1 PRA, and
- Be proficient in software development and interface issues.

These characteristics are key when considering potential avenues for this work.

At this point, the above work is being proposed, but is subject to the availability of the necessary resources as well as the approval of management.

10 CONCLUSION

A number of past studies have been reviewed for the purpose of characterizing the similarities and differences in the approaches employed. In addition, a synopsis of Level 2/3 regulatory considerations has been provided, including discussion of the relevant regulatory requirements, as well as the way that Level 2 and Level 3 models appear in the agency's PRA tools. Also, a review has been performed of advances that have been made in academia or elsewhere, but that are not reflected in the common (domestic) practice for Level 2/3 PRA. Following this, the feedback received from external stakeholders during a recent meeting to discuss Level 2 and Level 3 PRA modeling issues has been summarized. Next, four general Level 2/3 PRA approach categories have been outlined, for the purpose of allowing the comparison of an otherwise unlimited number of possible approaches. Subsequently, the basis for selecting

Approach 3 for future exploration has been articulated. Finally, possible variations have been investigated and future steps have been laid out.

APPENDIX A: REVIEW OF PAST STUDIES

This appendix provides a brief summary for a number of past Level 2 and Level 3 PRAs. It does not attempt to capture all details of the PRAs, only to discuss the general framework for the purpose of pointing out similarities and differences between the methodologies.

A.1 The Reactor Safety Study (WASH-1400)

The Reactor Safety Study (NRC, 1975), also known as WASH-1400 and the Rasmussen Report, sought to estimate the public risks that could be involved in potential accidents in commercial light-water reactor nuclear power plants. The approach taken (described in Appendix I of that report) is the basis for most subsequent PRAs with the Level 1 being handled via fault tress and event trees. For Level 1 sequences resulting in core melt, these sequences are mapped to containment event trees, as shown in Figure 10. Appendix VIII of WASH-1400 provides details of the containment failure mode analysis that forms the basis for the CET development described in Appendix I. Appendix VII of WASH-1400 provides the development of source terms associated with the various accident sequences that lead to core melt. Appendix VI of WASH-1400 provides the details of the consequence analysis that was performed.

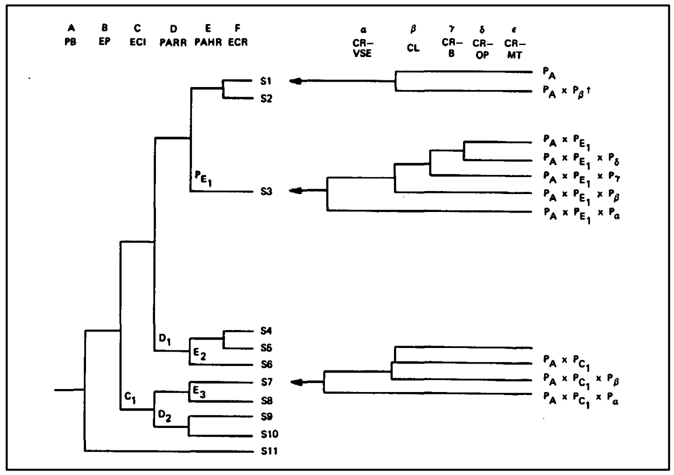

Figure 10: Illustration of WASH-1400 CET Mapping

The results of the Reactor Safety Study formed the basis for a subsequent update of reactor severe accidents source terms, documented in NUREG-0773 (NRC, 1982). While not a full Level 2, this study did represent a further development in treatment of accident progression, and utilized results from the Meltdown Accident Response Characteristics (MARCH) and Containment of Radionuclides Released After LOCA (CORRAL) codes.

A.2 Severe Accident Risk Estimates for Plants Licensed from 1981 – 1986 (Deferred)

Prior to 1980, Final Environmental Statements associated with initial plant licensing did not generally consider severe accident risk, based on the assertion that the probability of occurrence "is so small that their environmental risk is extremely low." (NRC, 1980) Following the Three Mile Island accident in 1979, the Commission released a policy statement (NRC, 1980) which required that initial licensing FES' consider severe accidents. In addition, SECY-81-25 (NRC, 1981) prompted additional studies for high-population sites.

In some cases, licensees or the NRC elected to develop site-specific source term frequencies and estimates to address severe accident risk, but more commonly the estimates were based on the results presented in NUREG-0773 (NRC, 1982). An accounting of the basis for the source terms of the plants in question (i.e., those licensed between 1981 and 1986) is provided in Table 5.2 of (NRC, 1996). These source term estimates were then used in conjunction with site-specific meteorology, population distributions, and emergency planning characteristics to calculate the airborne pathway environmental impacts.

Of particular interest from a historical methods standpoint are the analyses performed for the high populations sites (e.g., Indian Point), in that they were major undertakings and the first prominent Level 2/3 PRAs performed subsequent to WASH-1400. Unfortunately, time constraints prevented their summarization here.

A.3 The NUREG-1150 Study

The NUREG-1150 study (NRC, 1990) was conducted during the late 1980s and published in 1990. The study assesses the risks from severe accidents for five commercial nuclear power plants in the US. The study provides offsite risk results due to internal events for three sites and due to both internal and external events for two sites. Appendix B of NUREG-1150 walks through the entire assessment process for a specific station blackout event.

> Example PDS binning considerations:
>
> - RCS intact at onset of core damage
> - ECCS is recoverable
> - Containment heat removal is recoverable
> - A/C power is recoverable from offsite sources
> - RWST contents not injected, but could be if A/C power is recovered
> - TD-AFW failed at, or shortly after, accident initiation, MD-AFW is recoverable
> - RCP seal cooling is recoverable
>
> *[From NUREG-1150, App. B, pg. B-10]*

The Level 1 portion of the risk assessment is handled in a "traditional" manner utilizing fault tree/event tree analysis. Following quantification of the Level 1 model, which includes determination of success versus assumed core damage for each accident sequence, the plant damage states are defined. The plant damage states are created by grouping similar accident sequence cut sets, using the following criteria:

- The anticipated time to fuel uncovery
- Plant conditions at the time of fuel uncovery
- Anticipated accident progression characteristics after fuel uncovery

Practically speaking, cut sets from the same accident sequence are grouped in to the same plant damage state, but this is not necessarily always the case. The NUREG-1150 documentation describes this portion of the assessment as an iterative process performed by the accident frequency analysts and reviewed by the accident progression analysts. Plant damage states with a mean core damage frequency of less than $1 \cdot 10^{-7}$ were excluded from the accident progression analysis. (In the case of Surry, this meant that greater than 99% of the total mean core damage frequency was carried forward in to the accident progression analysis.) The remaining plant damage states (25 for Surry) were grouped in to PDS groups based on their initiating events (resulting in 7 PDS groups for Surry).

The 7 PDS groups represent the input to the accident progression analysis, which is achieved via the use of accident progression event trees. In the case of Surry, the APETs had 71 questions, or top events using the Level 1 nomenclature. Note that the APETs have so many top events that they are never actually depicted as event trees, but rather solved by a computer program (EVNTRE) as a series of questions. Since answers to some questions depend on the answers to previous questions, "cases" are defined in an attempt to capture these dependencies. Mechanistic computer codes (e.g., CONTAIN) and expert elicitation were used to (i) inform specific questions regarding quantitative or qualitative system characteristics and system responses at particular points in the accident progression and (ii) inform branch probabilities.

The results of the APET were then grouped in to accident progression bins. This was accomplished by EVNTRE using user-defined bin characteristics. In the case of Surry, there were 11 characteristics used to define the accident progression bins (APBs). An example of the characteristics for the Surry APET is shown in the box below.

> Example APET sequence binning characteristics:
>
> - Timing of containment failure
> - When containment sprays were in operation
> - Timing of core-concrete interaction relative to vessel breach
> - RCS pressure at vessel breach
> - High-pressure melt ejection?
> - Steam generator tube rupture?
> - Amount of core available for CCI
> - Amount of zirconium oxidized in-core
> - Amount of core involved in high-pressure melt ejection
> - Nature of containment failure
> - Status of RCS following vessel breach
>
> *[From NUREG-1150, App. B, pg. B-10]*

The APBs are the starting point for the source term analysis, which is performed using the XSOR series of codes (e.g., SURSOR in the case of Surry). The source term analysis provides the release fractions (for nine radionuclide classes) for early release and late release, additional information regarding release timing, the energy associated with the release, and the release height. The XSOR series of codes is not a mechanistic code. Rather, it builds upon the results predicted by earlier mechanistic calculations that supported the accident progression analysis, in conjunction with expert judgment. Distributions for key characteristics were developed and are probabilistically sampled by XSOR. This results in a very large number of source terms (~ 19,000 for Surry). As a result, these results are again binned, or in this case the term partitioning is more applicable. Since the release fractions of different radionuclide classes for a given source term are inherently coupled, it is not reasonable to treat them independently. Thus, "effect weights" were used to account for source terms that may contribute to both prompt and latent health effects versus those only expected to contribute to latent health effects. The partitioning process was accomplished using a computer code named PARTITION, and results in source term groups.

The STGs are the starting point for the consequence calculations, which were performed using the MACCS code. MACCS calculates the transport of the source term offsite, the subsequent dose received by the exposed population at numerous distances, and the resulting health effects. Dose pathways that are considered are cloud-shine, ground-shine, inhalation, and ingestion of contaminated food. One MACCS input deck corresponds to the source term of each STG. For each MACCS input deck, ~ 1,000 calculations are run to capture effects of weather variation. The final step taken is the combination of all the above results in the computation of risk, which is described in Section B.7 of NUREG-1150.

A.4 The LaSalle Study (Deferred)

A Level 3 PRA was performed for LaSalle Unit 2 as part of the Risk Methods Integration and Evaluation Program (RMIEP) and the Phenomenology and Risk Uncertainty Evaluation Program (PRUEP). This study is documented in NUREG/CR-5305 (NRC, 1992), but is not summarized here due to time constraints. This study was intended to be a more extensive exercising of state-of-the art methods than NUREG-1150, and therefore it would be advantageous to capture it as part of this overall methods survey at a future time.

A.5 Grand Gulf and Surry Low Power and Shutdown Studies

Studies were conducted for Grand Gulf and Surry to look at postulated severe accidents initiated during plant operation states other than full power operation and to compare the results to those from NUREG-1150 (NRC, 1995 & NRC, 1995a). For the Level 2/3 part of the study, only internal initiating events are considered. In the case of Grand Gulf the Level 2/3 portion focuses on accidents during operation mode 5 (approximately cold shutdown as defined at the time by the Grand Gulf Technical Specifications), and in the case of Surry the Level 2/3 portion focuses on accidents during mid-loop operation. The approach used for these studies is a simplification of the NUREG-1150 approach. Thus, only the differences are discussed here.

For the accident progression analysis, APETs are still utilized but there are fewer top events (questions). Specific phenomenological issues are dealt with in less detail in many cases, and formal expert elicitation is not used. For the partitioning of source terms in to source term groups, these studies utilized the modified-NUREG-1150 partitioning approach developed for the LaSalle study (Grand Gulf) or the PARTITION-LPS routine developed specifically for low power and shutdown (Surry). Offsite consequence calculations were handled identically to NUREG-1150, and these studies also included a scoping study of onsite consequences. It is worth noting that the offsite calculations were completed with a newer version of MACCS than that used in NUREG-1150. The newer version incorporated the results of the BEIR-V study (NAP, 1990).

A.6 Generic Environmental Impact Statement for License Renewal

NUREG-1437, Volume 1, the Generic Environmental Impact Statement for license renewal, includes a section (Chapter 5) on the risk from postulated severe accidents (NRC, 1996). No Level 2/3 analysis is performed. Rather, to assess the impacts from severe accidents, the 1996 GEIS relied on severe accident analysis provided in the Environmental Impact Statements (EISs) for the more recent plants (see Section A.2 for additional information).

The GEIS used the environmental impact information from the 28 plant-specific EISs and two metrics called exposure indices (EI), one for prompt fatalities and one for latent fatalities, to: (1) scale up the radiological impact of severe accidents on the population due to demographic changes from the time the original EIS was done until the year representing the mid-license renewal period and (2) estimate the severe accident environmental impacts for the earlier plants (whose EISs did not include a quantitative assessment of severe accidents). The EI method uses the projected population distribution around each NPP site at the middle of its license renewal period and meteorology data for each site to provide a measure of the degree to which the population would be exposed to the release of radioactive material resulting from a severe accident. To ensure that impacts were not underestimated, the GEIS EI methodology used the 95^{th} percent confidence interval of impacts for these plants. The EI metric was also used to project economic impacts at the mid-year of the license renewal period. A more detailed description of the EI method is contained in Appendix G of the 1996 GEIS (NRC, 1996). In this manner, the GEIS estimates severe accident risk using primarily previously developed information. However, the results of this evaluation are very conservative, and are designed for EISs, rather than best-estimate severe accident risk.

A.7 Individual Plant Examinations (IPEs)

Individually characterizing the Level 2/3 analysis performed as part of the full suite of IPEs is beyond the scope of this document. Instead, this section will summarize the relevant discussions from NUREG-1560 (NRC, 1997), which is a "rollup" of the IPEs. In addition, this section will outline the methodology used for a particular IPE (Palisades).

The IPEs varied in the methods and scope used for the accident progression analysis. This is attributed to the flexibility allowed within Generic Letter 88-20 and NUREG-1335 (IPE guidance documents). With regard to the definition of plant damage states, NUREG-1560 reports that some IPEs did not carry information (e.g., availability of power) forward through this Level 1/2 interface. In addition, in some IPE submittals the CDF was not conserved across this interface (PDS frequency noticeably less than CDF). This arose from one of three issues: (i) internal flooding sequences not carried forward, (ii) filtering of sequences that might not be insignificant to early release, and (iii) sequence filtering based on justifications that the Level 1 success criteria were too conservative.

The biggest optical departure from NUREG-1150 is the use of containment event trees, rather than accident progression event trees. The major distinction between the two is that containment event trees have significantly fewer top events. The majority of IPE submittals relied heavily, and in some cases exclusively, on reference plant (i.e., semi-generic) calculations for making determinations (e.g., CET split fraction values). The primary tool used for accident analysis was the MAAP code. NUREG-1560 indicates that licensees commonly did not address accident progression uncertainty via parameter distributions or sensitivity studies. NUREG-1560 also points to weaknesses in the treatment of operator action (particularly for PWRs) and consideration of severe accident environmental effects on equipment operability. Assumed mission times ranged from 24 or 48 hours to a time at which stable conditions had been reached. Lastly with regard to Level 2, NUREG-1560 includes criticisms of the treatment of source term analysis, from the standpoints of binning and release fraction characterization.

With regard to Level 3, NUREG-1560 notes that few licensees performed Level 3 calculations due to the nature of the IPE guidance, and that no sufficient review of these limited cases was performed to develop perspectives. For the licensees who did perform Level 3 calculations, either the MACCS or CRAC2 codes were used.

The Palisades IPE submittal (January 29, 1993) is used as a specific example of an IPE Level 2 analysis, as characterized in (NRC, 2008). It is not known to what extent this is a good "representative" example. Results from the Level 1 calculation are fed in to a bridge event tree to obtain plant damage states. Five characteristics were used in the bridge trees, as outlined to the right. The resulting number of PDSs was 392. To make the analysis more tractable, these PDSs were grouped based on the following:

> Bridge tree characteristics from Palisades IPE:
>
> - Initiator (e.g., large LOCA)
> - Time of core damage (e.g., early)
> - Secondary cooling? (e.g., available)
> - Pressurizer PORVs? (e.g., unavailable)
> - Containment systems? (e.g., containment sprays available and air coolers not available)

- Preemptive protective actions judged unlikely (i.e., all core damage timing assumed to be early) – reduced number of PDSs by 50%
- Some PDSs were judged to be illogical – reduced number of PDSs by 7%
- Truncation of sequences < $1 \cdot 10^{-9}$/yr – reduced number of PDSs by 25%[5]

[5] This value compares to the total CDF of $5.1 \cdot 10^{-5}$/yr. Note that later in the analysis, the frequencies of PDS below $1 \cdot 10^{-7}$/yr (which sum to less than 1% of the total CDF) are lumped in to the most severe PDS to further reduce the number of CET end-states.

- Removing splits based on PORV availability by taking a weighted average – reduced number of PDSs by 4%

The results is a total of 53 PDSs.

The next step in the Palisades IPE Level 2 analysis is the containment failure analysis. In this case, a finite element model was developed with underlying guidance form EPRI NP-6260-M and NUREG-1037. The analysis identified three areas for potential catastrophic failure and two types of locations associated with minor loss-of-pressure failure. From these independent failure projections, a composite containment fragility curve was developed. The net results was a projected failure pressure that is 2.4 times higher than the design pressure (131 psig versus 55 psig)

The Containment Event Trees were developed in conjunction with the PDSs, such that the PDSs contain only plant system information and the CETs address only the effects of severe accident physical processes. A common CET was used for each PDS to promote consistency in treatment and results. At this stage, the number of PDSs treated was further reduced by only carrying forward the top (from a frequency standpoint) 18 PDSs (see footnote 5). The frequency from the truncated 35 PDSs was added in to the PDS with the highest frequency. The common CET structure contained 12 top events, which were in turn informed by a substantial number of fault trees. The fault trees included ~ 100 basic events relating to PDS dependencies, recovery events, operator actions, and phenomena. The phenomena basic events comprised approximately ½ of the total number of basic events, and the probabilities for these basic events were varied depending on the situation.

The process for quantification was as follows: first the basic event probabilities are quanitified, next the fault trees are quantified, and finally the CET end-states are quantified. Since there are 18 PDSs and 65 CET end-states per PDS, this results in 18 x 65 end-states. To reduce the number of results, the corresponding CET end-state from each of the 18 PDSs are summed, resulting in 65 end-states. The licensee's version of the MAAP code was used to develop source terms for each of the 65 end-states (only 41 MAAP analyses were judged to be necessary).

A.8 Individual Plant Examinations: External Events (Deferred)

The post-Level 1 analyses associated with the IPEEEs represent the same challenge in summarizing as those for the IPEs, as described above. In the case of the IPEEEs, the overall summary document is NUREG-1742 (NRC, 2002), the counterpart to NUREG-1560. An investigation of the methods used for external events would be beneficial, especially if it included the evolution of those methods since the IPEEEs, however such an investigation was not practical within the constraints of the current effort.

A.9 License Renewal Severe Accident Mitigation Alternatives

As part of the environmental review performed for license renewal, licensees perform a severe accident mitigation alternative (SAMA) cost/benefit analysis, which effectively requires a site-specific Level 3 PRA. This analysis is reviewed by NRC and the results are included in the NRC's Environmental Impact Statement (EIS) for that plant's license renewal action. Collectively, these EISs form the Supplements to the Generic Environmental Impact Statement (NUREG-1437) discussed above. Both the licensee's submittals and the EISs are available on the NRC's external web site[6]. To date, license renewal applications at 26 sites have been approved, with 14 more applications under review.

The methods used by licensee's for performing these site-specific Level 3 analyses are not covered in detail here, because they vary, and there is currently no pre-existing compilation. However, they can be generally characterized as using methods for both Level 2 and Level 3 that are analogous (if not identical) to the methods used for the respective plant's IPE/IPEEE submittal, except for some cases where the simplified NUREG/CR-6595 (NRC, 2003) approach is used.

A.10 New Reactor Licensing

New reactor designs consider severe accidents as part of the licensing process. Chapter 19 of the Design Control Document provides this information, in two respects. As part of the safety review, the effectiveness of severe accident mitigation design features is deterministically assessed. In addition, to meet environmental requirements a cost/benefit analysis is performed to assess severe accident mitigation design alternatives (SAMDAs), which effectively requires a site-specific Level 3 PRA. Thus far, four designs have been certified by the NRC, with three more designs under review, as well as an amendment to one of the previously-approved designs. Here, we'll attempt to characterize the methods used for three of these designs: AP1000, US EPR, and US APWR.

The severe accident source terms developed in the design control document are also used in a combined license (COL) application to project offsite impacts for the specific site being considered. This is typically done using the MACCS2 code in a fairly standard manner, and is not covered here.

AP1000

For the AP1000 PRA (Chapter 19 of NRC, 2004), the end-states of the Level 1 corresponding to core damage are binned in to 11 accident classes based on the initiating event and the conditions at the onset of core damage. Each accident class (analogous to a plant damage state) is propagated through the containment event tree to evaluate the potential for operator actions, safety system response, and the containment structure to mitigate the release. Issues that are addressed by the CET top events are characterized to the right. CET split fractions are either linked to system-specific fault trees (for system-related top events) or assigned scalar values (for the other top events).

Issues covered by AP1000 CET top events:

- RCS depressurization after core uncovery
- Containment isolation
- Reactor cavity flooding (by gravity draining or manual actuation)
- Reactor vessel reflooding and associated hydrogen production
- Reactor vessel integrity
- Passive containment cooling
- Containment venting
- Intermediate containment failure
- Hydrogen igniter system availability
- Diffusion flames at IRWST and valve vault exits
- Early hydrogen detonation (during hydrogen release to containment)
- Global deflagration
- Intermediate hydrogen detonation (after hydrogen is mixed in containment)

[From NUREG-1793, Chapter 19]

6 http://www.nrc.gov/reactors/operating/licensing/renewal/applications.html

The CET end-states are in turn grouped in to six containment release categories:

- Intact containment
- Early containment failure
- Intermediate containment failure
- Late containment failure
- Containment isolation failure
- Containment bypass

Westinghouse conservatively assumes that all containment release categories not associated with an intact containment constitute a large release. Source terms were quantified using the MAAP4 code. These source terms were then fed in to MACCS2 for the Level 3 portion of the calculation, to obtain the effective dose equivalent whole body dose complementary cumulative distribution function at 0.8 km (0.5 mi) and the total person-rem exposure within 80 km (50 mi) of the plant. Since the analysis was done as part of a design certification application (i.e., not specific to a particular site), meteorology and population distributions developed by EPRI were used which bound 80% of the reactor sites in the US. Some sensitivity analyses were performed to look at the effect of uncertain elements of the analysis (e.g., effectiveness of external vessel cooling) on both the Level 2 (e.g., LRF) and Level 3 (e.g., population dose) results.

US EPR

For the US EPR, core damage sequences are grouped in to 30 groups, described in Table 19.1.16 of (Areva, 2007), and termed Core Damage End States (CDESs). Based on the characteristics of the CDESs, CDES link event trees link each CDES to 1 of 7 differing containment event trees, described in Table 19.1-18 of that document. An eighth CET is also used, but is only called from within one of the seven CETs mentioned above. The number of top events in the various CETs range between 1 and 13.

For the EPR, linked fault trees are used for the Level 1 and Level 2 models, meaning that the same system fault trees that form the underpinning of the Level 1 model are utilized for the Level 2 model. The advantage of this approach is that the Level 1 and 2 interface is handled more directly, without the need to duplicate top event information. For phenomenological top events, either fault trees or decomposition event trees are used.

> Characteristics used in release category grouping:
>
> - Containment bypass
> - Timeframe for containment failure
> - Containment failure category
> - Melt retained in-vessel?
> - MCCI?
> - Melt flooded ex-vessel?
> - Source terms mitigated by sprays/scrubbing?
>
> *[From (Areva, 2007), Section 19.1.4.2.1.3]*

Containment fragility curves were developed for the six dominant containment failure modes (cylinder wall, center of dome, base of cylinder, base of dome, equipment hatch V2, and equipment hatch H2). Median failure pressures ranged from 180 to 280 psi (see Table 19.1.21 of Areva, 2007). These individual fragility curves were combined in to a composite fragility curve for use by the Level 2 PRA analysts.

To facilitate construction of the CETs and later binning, three timeframes are defined. Using these timeframes in the context of the relevant accident characteristics (described above), the resulting CET sequences are grouped in to fission product release categories. A total of 25 release categories were used, as described in Table 19.1-19 of (Areva, 2007).

Based on the result of the release category binning described above, calculations were run using the MAAP4.0.7 code to develop representative source terms. Sensitivities were run to investigate the effects of (i) isolation failure size, (ii) importance of the Severe Accident Heat Removal System on the source term, (iii) Safeguards/Fuel Building holdup for ISLOCAs, and (iv) water pool scrubbing during SGTRs. The definition of large release is adopted from NUREG/CR-6595, relating to the type of containment failure (i.e., bypass sequences) and the release fraction of I, Cs, and Te (i.e., greater than roughly 3%). Note that for new reactors, the relevant metric is LRF rather than LERF.

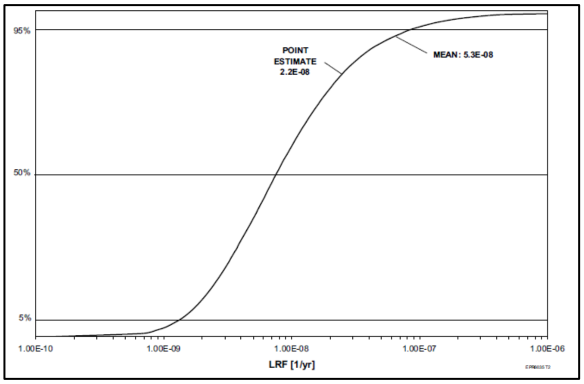

Figure 11: US EPR LRF Uncertainty Distribution

An uncertainty analysis was performed which assigned probability distributions to system fault tree basic events, human actions, and phenomenological basic events. The resulting uncertainty distribution for LRF is shown in Figure 11 (Figure 19.1-9 of Areva, 2007).

US APWR

For the US APWR, the methodology used is documented in (Mistubishi, 2008) and (Mitsubishi 2008a). Section 19.1.4.2 of the former document describes the Level 2 PRA for operations at power beginning with the accident class (ACL) logic model described in a previous section (19.1.4.1.1). The failure end states of the Level 1 PRA event trees result in ACLs that are initial conditions for the Level 2 analysis. In total, 28 ACLs are defined for the US APWR PRA. The logic tree for ACL classification is shown in Figure 19.1-6 (Mitsubishi, 2008). The set of 28 accident classes is used to start the Level 2 quantification process. Linking and quantification of the Level 1/2 models are performed using the Risk Spectrum code.

Classification of ACLs:

- Initiating events and primary system pressure
- Containment intact or failed at core damage
- Accident progression in containment
- Loss of support system initiating events

[From (Mitsubishi, 2008), Section 19.1.4.2]

The CET development considers (i) containment failure timing, which in turn affects the characteristics of fission product release to the environment, (ii) important phenomena in containment that may cause containment failure, and (iii) recovery of safety system and accident management operations that may contribute to preventing containment failure. The CET actually consists of two portions, the Containment System Event Tree (CSET) and the Containment Phenomenological Event Tree (CPET). The interface between the CSET and CPET is defined to be the plant damage state, which forms the end states of the CSET and the initial conditions of the CPET. In this respect, the CSET fulfills a similar function to bridge event trees in other methods.

In total, 72 PDS are defined for the US APWR. These PDS are classified taking into consideration:

- Primary system pressure
- Reactor cavity flooding status
- Containment status at core damage
- Igniter status
- Containment spray system status
- Containment heat removal status

The conditional probability of each CET end state for each PDS is quantified by spreadsheet models of the CPET. Failure fractions of the top event of the CPET are quantified using one of the following methods:

- Quantification by applying the results of PRAs previous to the US APWR PRA,
- Quantification by analyzing the load due to the physical phenomena of concern in comparison to the structural capacity, or
- Quantification by substituting the qualitative evaluation results according to the accident progression analysis by the MAAP4 code with examination of the knowledge of severe accident phenomena and evaluation examples in previous PRAs.

Source terms were quantified using the MAAP4 code.

The MACCS2 code was used for the Level 3 PRA analysis to estimate the population dose for each release category source term. The meteorological data of the Surry site was used as "typical." The 50-mile population data of the Surry site in the MACCS2 code sample input file was adjusted to be representative of about 80% of the U.S. nuclear plant sites.

APPENDIX B: RECENT DOMESTIC AND INTERNATIONAL DEVELOPMENTS

This appendix provides a brief summary of some pertinent domestic and international developments in the way that one might approach Level 2 and/or Level 3 PRA. The appendix is not intended to be comprehensive.

B.1 NRC WinNUCAP / SAPHIRE Level 2 Models

The models discussed below are very similar to those discussed in Appendix A. They are discussed in this appendix because they (a) represent the agency's most recent Level 2 model development effort, and (b) have not been utilized to conduct an actual Level 2 PRA to date, other than comparison back to the licensee's Level 2 PRA.

As part of an exploratory effort, INL and BNL (under NRC contract) developed modified versions of a few licensee's Level 2 PRA models[7]. These efforts supported an initiative to upgrade the capability of Level 2 modeling in the agency's SPAR models, which is discussed elsewhere in this report (Section 3.2). The models utilized extended Level 1 event trees (i.e., transfer event trees which mimic additional top events, beyond core damage, to identify the availability of systems relevant to accident progression). The Level 2 portion of the model was developed, based heavily on licensee models, in the WinNUCAP software to take advantage of that software's handling of DETs. Later, the models were converted to the agency's PRA software (SAPHIRE 7) using logic rules to replace the DETs. Thus, in the final SAPHIRE 7 models, DETs are not directly used in the quantification process.

For the WinNUCAP models, the input to the Level 2 is the end-states of the extended SPAR Level 1 event trees. Since the BNL effort was a feasibility study, and since transfer of sequences from the SAPHIRE-based extended Level 1 event trees to WinNUCAP has not been automated, a subset of the accident sequences are carried forward. For the three models developed, these subsets accounted for 88% - 97% of the respective CDF. The subset of sequences is then binned in to plant damage states using logic rules (which can be represented as an event tree) based on system availability and expected accident progression characteristics. In the case of all 3 WinNUCAP models developed, the PDS trees had 11 top events.

The resulting plant damage states are then fed in to a containment event tree. For the three models developed, the number of top events ranged from 7-10. Each top event has a unique supporting decomposition event tree. The number of top events in the DETs ranged from 1 to 9. The source term category event trees for the three models had between 6-7 top events, based on the 6-7 CET characteristics that were deemed important to the source term.

As with the BNL models, the starting point for the INL Level 2 models is the extended Level 1 SPAR models. The extended models use containment system transfer event trees to mimic the inclusion of additional top events relating to containment systems in to the Level 1 event tree. The resulting sequences are then binned using SAPHIRE's partitioning capability, based on 11 characteristics, which form a PDS event tree. Each sequence carries forward an 11-character PDS identifier which contains the relevant system availability and sequence characteristic information. These results are then fed in to a containment event tree. The end-states of the CET are then binned in to source term categories via a source term category event tree.

---

7 Models were developed for Surry, Peach Bottom and Sequoyah.

An example of the implementation of this type of approach (extended Level 1 event trees in conjunction with DET-supported CETs) internationally can be found in Kovacs, 2008. Also, others have proposed very similar model structures that contain most, if not all, of the same basic features as described above (e.g., see Torri, 2009).

B.2 Bayesian Networks

Though not new, use of Bayesian Networks (or Bayes Nets) has gained increasing attention. Bayes Nets are models of a particular “world” that provide the framework for assigning causality and probabilities between a large number of inter-related items. For instance, a Bayes Net could be built to describe the behavior of a system based on its individual components and the external factors that would affect it (e.g., environmental qualification considerations). In this context, the Bayes Net would fit a very similar function as a fault tree, and could be used in a traditional (event tree-driven) Level 2 approach in place of fault trees or DETs. In this context, where a CET is employed with underlying models corresponding to each top event, but with those underlying models being quantified prior to event tree quantification, a mixture of fault trees, DETs, and Bayes Nets could be used.

At a higher level, Bayes Nets could also be used in lieu of an event tree. However, it is not obvious that this is tractable for a system as complex as an NPP, or that it has inherent advantages over an event tree in terms of its ability to better integrate with phenomenological tools.

B.3 Circa 2002 NRC/SNL/LANL Interactions

The question of whether Level 2 modeling should be evolved, and how this should be accomplished, is not a new one. In addition to the efforts described in Sections 3.2 and B.1, the NRC had limited interactions with SNL and LANL in the 2002 timeframe on this issue (McClure, 2002). During these interactions, the NRC proposed exploration of a sampling-based simulation method, which was termed “Probabilistic Physical Process Modeling.” This approach uses a driver program to randomly sample from probability distributions for all key system code input. Next, the system code would be executed to quantify the result for that particular set of inputs, and the result would be placed in to a source term bin. This process would be repeated as many times as necessary to arrive at a stable distribution of results (meaning that additional samples don’t significantly change the results). The major weaknesses of this approach, as identified by SNL and LANL, are:

- The number of calculations required is impractical.
- System codes (in this case MELCOR) may not have all of the physical models needed to be applied this generally.
- The approach, as proposed, doesn’t account for uncertainties in the accident progression itself (e.g., uncertainties as to when hydrogen combustion will occur).

Based on these perceived shortcomings, SNL and LANL proposed a compromise approach which attempted to incorporate the strengths of both the traditional modeling and the physical process modeling. Their proposal is characterized as follows:

1 An event tree is constructed to assign probabilities to scenarios. Rather than constructing the event tree to reflect the cumulative knowledge of accident progression, the top events would be selected to reflect important MELCOR boundary conditions.

The proposal indicates that “signature events” (e.g., failure of an SSC) would form the basis for branch selections, such that sequences can be more readily binned. Similar sequences could be grouped, and the event tree would be used as an importance-sampling device for identifying the key MELCOR calculations needed to scope out the relevant accident progression.

2 MELCOR is used to calculate source terms, by running a calculation for each source term bin and additional calculations to address uncertainties.

3 The results from the above steps are then used to iteratively refine the event tree to ensure that the MELCOR calculations are capturing the most important features of the accident.

Due to significant redirection of the project that prompted the above interaction, significant progress was not made toward either of the two proposals.

B.4 Treatment of Uncertainties in Thermal Hydraulic and Severe Accident Analysis

One major aspect of Level 2 analysis is the treatment of numerous accident sequences due to the uncertainties associated with operator response, system availability, and accident progression. As such, there is commonality between this aspect of Level 2 analysis and the treatment of parameter and modeling uncertainty in thermal-hydraulic and severe accident analysis. This latter area has received significant attention over the past twenty years, and a few particularly relevant activities are highlighted.

Historically, a number of studies to develop methodologies which (amongst other things) would account for deterministic analysis uncertainty were undertaken in the late 1980s and early 1990s (e.g., (NRC, 1989) relative to Code Scaling, Applicability, and Uncertainty – CSAU; (NRC, 1991) relative to Severe Accident Scaling Methodology – SASM). Work in this area has continued in the 1990s and 2000s, including further development and application (e.g., see (NE&D, 1998)) and numerous international workshops and studies (e.g., (CSNI, 1994), (NRC, 1997), (CSNI, 1998), (CSNI, 2005)).

One particular study of note involving quantification of uncertainty for a particular severe accident phenomena (hydrogen generation) is documented in (SNL, 2003). In this study, selected model parameters were randomly sampled using latin hypercube sampling to develop a distribution of results for the mass of hydrogen generated. For this study, 40 simulations were performed. An executive program (a modified version of DesktopPA) was utilized to automate the generation of input and perform limited output processing. An example of the results produced for in-vessel hydrogen generation is shown in Figure 12.

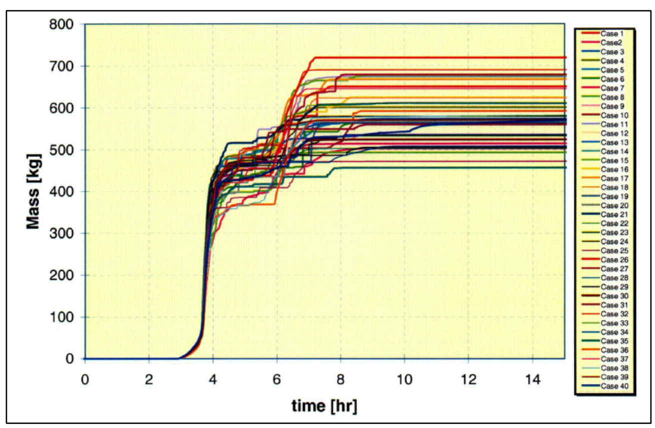

Figure 12: Sample Results of Hydrogen Generation Uncertainty

Finally, two methodologies that are currently being developed and/or applied under the auspices of CSNI are the SM2A (Safety Margins Applications and Assessment) project which deals with the change in safety margin and risk associated with a given plant change and the BEMUSE (Best Estimate Methods, Uncertainty and Sensitivity Evaluation) project which deals with the use of probabilistic methods for estimating result uncertainty associated with thermal-hydraulic simulations.

B.5 ADS-IDAC

The Accident Dynamics Simulator with the Information, Decision, and Actions in a Crew context cognitive model (ADS-IDAC) is a simulation-based approach developed at the University of Maryland to couple a thermal-hydraulics program (RELAP5) with a module for addressing operator interactions with the system (see Chang, 2008 and Coyne, 2008). The methodology uses a discrete dynamic event tree (DDET) approach by applying simple branching rules to account for variations in crew responses to plant events and system status changes. Since the branching points are dynamically generated, each sequence can have a unique set of top events when visualized as an event tree. However, it is possible to simplify the DDET to appear more like a conventional event tree (see Figure 13).

Examples of branching rules used within IDAC include the following:

- Use (by the operator) of memorized information versus always consulting the control panel;
- Use (by the operator) of knowledge-based actions versus implementation of the emergency operating procedures;
- The activation of a mental belief (on the operator's part) versus that mental belief remaining dormant;
- Equipment failure versus equipment failure with subsequent recovery versus unrecoverable failure;
- Inadvertent skipping of a procedure step;
- The amount of time that a particular action takes.

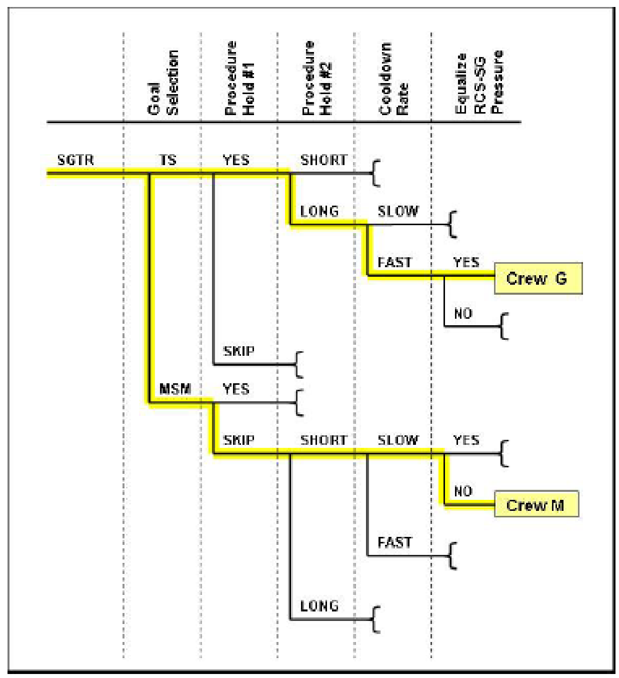

Figure 13: Example ADS-IDAC Simplified DDET

In the context of the current effort, the two key features of the approach are that (a) it is most useful in studying a particular scenario due to the computational and problem setup requirements, and (b) it is designed to investigate human interaction effects (though more traditional events such as hardware failure can be integrated in to the framework). Multiprocessor capability is employed to increase computational efficiency.

B.6 ADAPT/MELCOR

The ADAPT/MELCOR interface developed at Ohio State University combines a dynamic event tree generator (ADAPT) with an accident progression computer code (MELCOR), see (Hakobyan, 2008). The main focus of this work is to address the semi-static limitation associated with traditional event tree approaches (i.e., that top events in an APET/CET loosely imply a sequence of events based on practitioner judgment and decoupled computer code analysis).

ADAPT is designed to handle both active and passive system performance, and both aleatory and epistemic uncertainty. In the context of ADAPT, elements that are treated stochastically include:

- behavior of active components (e.g., probability of a valve failing on demand)
- behavior of passive components (e.,g., probability of a steam generator tube spontaneously rupturing)
- phenomena (e.g., probability of hydrogen combustion)

Conversely, modeling characteristics (e.g., heat transfer coefficients) are treated as epistemic. In this context, ADAPT uses user-assigned probability density functions for sampling the stochastic elements. This means that rather than having a particular event (e.g., hydrogen deflagration) occur at a specified set of conditions, the user specifies a range of conditions (and associated probabilities) for which the event could occur, and ADAPT randomly samples to determine when the event actually occurs (i.e., when a branching point is needed).

> Severe accident phenomena that can be stochastically defined and sampled in ADAPT/MELCOR:
>
> - Creep rupture of major RCS components (e.g., surge line)
> - Hydrogen combustion
> - Containment failure due to overpressure
> - Sticking open of pressurizer relief valves
> - Power recovery (for station blackouts)
>
> *[From (Hakobyan, 2008)]*

The generation of event trees within ADAPT is handled by a driver that

1. manages branching rules (i.e., determines when a branch point is needed);
2. handles system code initiation, termination, and file processing;
3. determines scenario probabilities; and
4. combines similar scenarios (based on user criteria and system code results).

Like other dynamic event tree approaches, ADAPT/MELCOR utilizes multi-processor capabilities to increase computational efficiency. Hakboyan, 2008 provides an application of the ADAPT/MELCOR approach to investigate the induced steam generator tube rupture issue.

B.7 MCDET

The probabilistic dynamics method MCDET (Monte Carlo Dynamic Event Tree) ((Kloos, 2006), (Peschke, 2008)) is a combination of Monte Carlo (MC) simulation and the Discrete Dynamic Event Tree (DDET) method. This method was developed at the Gesellschaft für Anlagen und Reaktorsicherheit (GRS; Germany). It was implemented as a stochastic module which can operate together with any deterministic code (e.g., MELCOR; ATHLET) simulating the dynamics of the underlying system. The combination is capable of handling the interactions over time between stochastic influences as specified in MCDET and the dynamics as modeled in the deterministic code.

> MCDET Approach:
>
> - Capable of accounting for any deterministic or stochastic event.
> - Events associated with a time and system change (deterministic or random) are treated by the DDET.
> - Events for which the timing and/ or system state change are random and continuous are treated by Monte Carlo.
> - Events with discrete alternatives may also be represented by MC simulation.

Using Monte Carlo sampling, multiple random sets of conditions are obtained for those aspects of the analysis that correspond to events for which the timing and/or system state change are random and continuous (generally corresponds to stochastic effects). Each set of conditions corresponds to a single discrete dynamic event tree. The DDETs in turn treat events for which the timing and/or system state change are either (i) deterministic or (ii) random and discrete. The calculation of each sequence is controlled such that sequences with a probability less than a user-defined threshold value are terminated. In addition, the branching is handled such that sections of the tree shared by different sequences are only computed once.

In addition, a Crew Module has been developed to couple with MCDET. This module is able to simulate a procedure of operator actions as a dynamic process, including stochastic elements (e.g., the execution time of a particular action) and deterministic elements (e.g., exchange of information through communications). This is accomplished through a set of user-prescribed lists (scripts) and program routines. However, the module does not attempt to model the mental process and cognitive behavior of the crew. In combination with MCDET, the Crew Module handles the time-dependent interactions between the human actions, system behavior, and stochastic events.

B.8 Other Level 2 Developments

A number of other related approaches have been developed in the past, and some form the basis for the approaches described above. For a summary of some of these approaches (dynamic logical analytical methodology – DYLAM; dynamic event tree analysis method – DETAM; dynamic event network distributed risk-oriented scheduler – DENDROS) see (Hakobyan, 2008).

A recent Level 2 PRA carried out by IRSN is described in (Raimond, 2008), which uses some variations on the standard NUREG-1150 approach in quantifying the effect of SAMGs. While termed a dynamic event tree approach, it is different from the approaches described above. An "expanded" Level 1 / Level 2 interface exists, which results in significantly more PDSs. Results from representative ASTEC calculations are used to define the amount of key parameters (e.g., hydrogen mass) before and after specific top-level events in the APET. Operator action timing / success probabilities are also based on these representative calculations, and quantified in the APET.

Also of note is a soon to be released paper (Mercurio, 2009) which investigates clustering of sequences that have been developed by a dynamic event tree tool. In the example, the authors from PSI and Polytechnic of Milan use the ADS-IDAC tool to investigate the steam generator tube rupture issue.

B.9 Current Handling of Uncertainties in MACCS2

Historically, consequence analysis has always considered weather variability as a key variation in the consequence model. For decades, consequence analysis codes have had the capability to perform sampling of actual weather data to address this issue, and this continues to be the case in MACCS2, which allows for stratified sampling and stratified random sampling. However, MACCS2 also allows for treatment of parametric uncertainty, including the parameters that describe the source term input. Most parameters in a MACCS2 input model can be prescribed uncertainty bands via user-specified ranges of the value in conjunction with degrees of belief across that range. For parameter uncertainty, MACCS2 offers the user a choice of functional forms, with a look-up table capability also being available.

In addition, correlation coefficients can be prescribed to define the constraints between various parameters. Practically speaking, developing these correlation coefficients is very difficult for source term parameters. In practice, the most straight-forward approach to considering source term uncertainty in MACCS2 is to input a set of n, equally likely, source terms (where n is an integer). Each MACCS2 realization would then use: (1) the exact same weather sequences, chosen to sample the meteorology file; (2) one of the n source terms; and (3) a randomly chosen set of other-than-source-term paramters. The total number of MACCS2 realizations is user-chosen, but is adjusted upward by the code, if necessary, to be an even multiple of n, so that each source term is used the same number of times.

APPENDIX C: SUMMARY OF MEETING WITH TARGETED EXTERNAL STAKEHOLDERS

On March 9th, 2009, a number of invited experts representing a broad range of interests gathered in person, or via phone, to discuss issues related to modeling in Level 2 and Level 3 PRA. The purpose of the meeting was to gather input for an ongoing NRC activity related to this subject area. The following sections provide some accounting for the ideas discussed at the meeting, without attempting to create a meeting record or attribute specific views to specific participants. A participant list is provided at the end of this appendix.

Level 2/3 framework

- As expected, a number of different views were expressed regarding the preferred approaches for Level 2/3 conduct. Even for traditional approaches, there was no consensus on the treatment of issues such as the definition of plant damage states.
  - As an example, there was significant disagreement as to whether systemic and phenomenological events can be physically separated in the event tree.
- Several participants noted the benefits of simulation-based tools in conduct of a Level 2/3 PRA and/or training.
- Several participants noted that the number of PDS that can be handled, when PDS are used, is much higher than what is routinely done.
- Similarly, several participants felt that a much larger number of release categories could be tolerated computationally, thus requiring less binning (and associated approximation).

Treatment of EOPs / SAMGs / EALs

- There were widely varying views as to the extent to which existing PRAs adequately address human response. Even so, most (if not all) participants felt that there were aspects of operator response that could be improved.
  - Many participants agreed that the synchronized tracking of EPs/SAMGs with the accident progression simulation is important. Network models, rule-based system code inputs, and cognitive models were all mentioned.
- The difficulties in modeling human response and human error for SAMGs was noted, given that SAMGs are not prescriptive, and not always deterministic. Some participants viewed accurate human error probability quantification as a significant technology gap.
- Several related points that were raised include:
  - It was noted that the modeling of recovery actions relates to the Level 1 treatment as well.
  - Work at Westinghouse was cited as indicating that only a handful of post-core damage operator actions have a significant effect on Level 2 results[8].
  - It was noted that one insight from the SOARCA project is that all sequences studied are well-suited for mitigation.
- Regarding EALs, the importance of warning time on the Level 3 results was cited. One participant contributed that variability exists in the declaration of a general emergency too early, but otherwise, there is fairly consistent treatment of the EALs across industry.

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[8] R. Lutz and M. Lucci, *Modeling Post-Core Damage Operator Actions in the PRA*, ANS PSA 2008 Topical Meeting, Knoxville, TN, September 7-11, 2008.

Collectively, the group believed that the accurate modeling of EAL effects is not an important limitation of existing methods.

Treatment of uncertainty

- Again, significant differences existed amongst the participants' views on the specifics of treating uncertainty, though most seemed to agree that a better job could be done at quantifying the parametric and modeling aspects. It was noted that this issue has come up repeatedly in the discussions of the ANS Level 2 standard writing group.
- Examples where fundamental differences existed involve the view of structural and phenomenological event uncertainty. Whereas some participants feel strongly that these uncertainties are entirely epistemic, others feel strongly that they are stochastic.
- Relating back to the framework discussion above, many participants felt that the preferred structure of the Level 2 PRA is intrinsically linked to the philosophical view of uncertainty. The treatment of uncertainty was used both as the basis for why systemic and phenomenological events can be separated and for why they can not.
    - As stated by one participant, the NRC should decide what view of uncertainty it wants to propagate prior to endorsing a particular approach.
- There was also significant diversity in the perception of the importance of phenomenological uncertainty, ranging from those who view it to be very significant to those who view it to be greatly exaggerated.
    - It was noted that many Level 2/3 PRAs focus on uncertainties that explicitly affect release/consequence frequencies (akin to the way uncertainties are treated in a Level 1 PRA), without focusing on the underlying parametric and model uncertainty. Some felt that this treatment is a significant limitation of existing methods.
    - It was noted that the SOARCA project is undertaking an uncertainty quantification study in an attempt to estimate the importance of these uncertainties.
- Several participants felt that any research effort undertaken should view the informing of uncertainties as a primary objective. Specifically, such an effort could act to direct resources in to areas where uncertainty is high, while identifying other areas where simplified approaches are justified.

Scope and fidelity of the PRA model

- Almost all participants seemed to feel that the fidelity of existing models could be improved, in one way or another. There was no consensus as to how this should be achieved.
- Several participants noted that completeness and fidelity are natural enemies of one another, due to resource limitations.
    - Several comments were geared toward the need for better guidance for the practitioner as to which aspects of the analysis required more rigor, versus which were amenable to simplification (similar to comment above under *Uncertainty*).
    - Related to guidance, some participants felt strongly that past NRC decisions (e.g., treatment of DCH) were very important in limiting the analysis scope, and thus allowing for more attention to fidelity.
- Some participants felt that greater fidelity could be achieved by methodological changes (e.g., use of dynamic modeling) while others felt that the state-of-knowledge justifies the coarse treatment of phenomena in the existing methods.

- One case was cited (AP1000) where additional work resulted in better scrutability, but no change in the overall conclusions.
- Conversely, preliminary SOARCA insights were characterized as demonstrating that accidents progress much more slowly and result in much more modest releases as compared to recent (and past) Level 2/3 PRAs.

- It was noted that the desired fidelity of a model depends on the results usage (see *Results* below), as well as whether the analysis purports to model a single plant or a class of plants.
- Regarding fidelity, one participant offered that the litmus test for a model's fidelity should be whether it accurately reflects the events that would transpire at the actual plant.
- Regarding Level 3 modeling, there was general agreement that more rigorous models exist (e.g., variable-trajectory models) for many aspects of the Level 3 analysis, but that in some cases these improvements would not result in a change in the final results.
  - ATD models were cited as an example where greater fidelity might not overly affect the results. However, it was noted that there are plans to upgrade this model in MACCS2 regardless, due to the attention that it receives.
  - Conversely, the effect of non-nuclear offsite consequences from large seismic events was noted as an area that would significantly benefit from additional attention. Similarly, upgrades needed to look at alternative risk metrics was raised (see *Results* below).
- The ability to handle more sensitivity studies for long-term recovery and more source terms than is routinely modeled was raised. The latter point is important from the standpoint of improving the un-quantified loss of fidelity associated with release category grouping.

Reliance on the practitioner versus reliance on computational tools

- As alluded to above, there was a wide range of views expressed relative to the appropriate reliance on computer simulations, ranging from those who view it as an integral and essential part of the framework itself, to those who believe it should simply inform the practitioner. Others have a more centrist view, where the computer simulation greatly informs the treatment of phenomena, while the practitioner provides judgment as to which aspects of the analysis receive detailed treatment versus which can be treated simplistically.
- One sentiment expressed is the importance of ensuring that Level 2/3 practitioners stay up to speed with developments in severe accident research, such that all major phenomena are appropriately addressed.
  - Several participants made the case that integral severe accident codes are an important mechanism for providing consistency between deterministic severe accident analysis and Level 2/3 PRA.

Use of results

- A repeated point made by multiple participants during the meeting was the importance of defining the use of the results:
  - For Level 2 PRAs, are intermediate risk metrics desired versus the full results needed to conduct a Level 3 PRA? (The organizer clarified that this activity assumes the conduct of a Level 3 PRA.)

- For Level 3, are only prompt and latent health effects desired, or are alternative risk metrics (e.g., land contamination) also of interest? (Again, it was clarified that this activity plans to at least consider alternative risk metrics.)

- Points related to this that were brought out by participants included:
  - Biasing of results to ensure conservatism often results in the mis-portrayal or mis-use of those same results elsewhere as best-estimate. This can act to perpetuate undue conservatism in accident management and EP, and perpetuate an out-dated view of risk-significance.
  - There has been increased interest recently by others (e.g., other federal agencies) in risk results other than health effects (i.e., economic and environmental impacts).
  - Design changes and decreases in severe accident phenomenological uncertainty have driven down large early releases, thus increasing the relevance of other metrics (e.g., land contamination).
  - The risk-significant features of the Level 2 results are dependent on the end-use (e.g., release timing is important for prompt health effects, but not for land contamination).

High-Level Takeaways

It was not the intent to reach consensus on any specific issues as part of this brainstorming session. Even so, there are certain issues where some level of agreement was reached. These include:

- Binning effects at both the front-end and back-end of the Level 2 analysis can be reduced by the treatment of more PDS (when used) and more release categories.
- The treatment of operator response (including HRA) and accident management in Level 2/3 PRA would benefit from additional modeling attention. The tracking of EOPs/SAMGs synchronously with accident progression is beneficial.
- The treatment of parametric and modeling uncertainty can be improved, and in the long-term would benefit practitioners in terms of focusing attention on uncertain risk-significant aspects of the analysis.
- The view of uncertainty fundamentally affects the preferred construction of a Level 2 methodology.
- Given the desire to use Level 2/3 results for an application other than regulatory risk metric compliance, the fidelity of existing models could be improved.
- Resource limitations will always necessitate choices between completeness and fidelity. These limitations should be recognized and research should aim to improve guidance on where simplification is acceptable versus where it is not.
- The running of physical models (e.g., MELCOR) continues to be an important aspect, in terms of capturing non-intuitive effects and ensuring that the Level 2 PRA does not become too out-of-step with severe accident research
- Specific Level 3 modeling issues could be informed by more attention (e.g., non-nuclear seismic consequences, alternative risk metrics).
- The desired end-use of the results should fundamentally affect the intermediate modeling decisions.

Other notable points where a partial consensus was not reached include:

- The view of phenomenological and structural response uncertainty (and their associated treatment in split fractions) as either primarily stochastic or epistemic.
- The specific approach that should be taken in improving Level 2/3 PRA model fidelity.
- The degree to which upgrades in most Level 3 aspects (e.g., ATD modeling) would affect the end results of interest.
- The view of how much reliance should be placed on the practitioner versus a computer simulation.

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