Uncertainty Analysis for the U.S. NRC State-of-the-Art Reactor Consequence Analyses

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Abstract: The U.S. Nuclear Regulatory Commission (NRC) recently published its draft State-of-the-Art Reactor Consequence Analyses (SOARCA) reports on the possible public health consequences of severe U.S. nuclear power plant accidents that could release radioactive material into the environment. SOARCA takes advantage of national and international reactor safety research and reflects updated plant design, operation, and accident management implemented over the past few decades. The overall objective of the SOARCA project is to develop a body of knowledge on the realistic outcomes of severe reactor accidents using self-consistent, integrated modeling of accident progression and offsite consequences drawn from current best practices modeling to estimate offsite consequences for important classes of events. This paper reports on the SOARCA uncertainty analysis that is in progress, for the long-term station black-out scenario for the Peach Bottom boiling-water reactor pilot plant. The goals of the uncertainty analysis are to develop insight into the overall sensitivity of the SOARCA results to uncertainty in key modeling inputs, identify the most influential input parameters for releases and consequences, and demonstrate uncertainty analysis methodology for severe accident consequence analyses and level 3 probabilistic safety assessments. Key uncertain parameters in both the MELCOR model for accident progression and radionuclide release (the level 2 modeling) as well as the MACCS2 model for consequences to the public (the level 3 modeling) are varied. A variety of techniques are being used to analyze the results, including partial rank correlation coefficients, stepwise rank regression analyses, scatter plots, and inspection of individual Monte Carlo realizations of interest. This paper presents the methodology for the integrated uncertainty analysis and the model parameters selected for this uncertainty study.

1. INTRODUCTION

This paper describes the NRC's uncertainty analysis for the SOARCA project's "unmitigated" long-term station blackout (LTSBO) severe accident scenario at the pilot boiling-water reactor, Peach Bottom.

The overall objective of the SOARCA project is to develop a body of knowledge on the realistic outcomes of severe reactor accidents utilizing self-consistent, integrated modeling of accident progression and offsite consequences drawn from current best practices modeling to estimate offsite consequences for important classes of events. This was accomplished by deterministic best estimate modeling of accident progression (i.e., reactor and containment thermal-hydraulic and fission product response), which is embodied in the MELCOR code, coupled with modeling of offsite consequences in the MELCOR Accident Consequence Code System, Version 2 (MACCS2) code in a consistent manner and with improved input in important areas. The uncertainty analysis is a focused study of the epistemic (state-of-knowledge) uncertainty associated with the accident progression and offsite consequence modeling.

1.1. Background

The evaluation of accident phenomena and offsite consequences of severe reactor accidents has been the subject of considerable research by the NRC, the nuclear power industry, and the international nuclear energy research community. In the past few decades, numerous influential changes have occurred in the training of operating personnel and the increased use of plant-specific capabilities. Plant changes and improved understanding from research include the following: (1) the transition from event-based to symptom-based emergency operating procedures (EOPs); (2) the performance and maintenance of plant-specific probabilistic risk assessments (PRAs) that cover the spectrum of accident scenarios; (3) the implementation of plant-specific, full-scope control room simulators to train operators; (4) an industry wide technical basis, owners-group-specific guidance, and plant-specific implementation of the severe accident management guidelines (SAMGs); (5) additional safety enhancements, described in Title 10, Section

50.54(hh) of the Code of Federal Regulations (10CFR50.54(hh)). These enhancements are intended to be used to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities under the circumstances associated with loss of large areas of the plant due to explosions or fire, to include strategies in the following areas: (i) firefighting; (ii) operations to mitigate fuel damage; and (iii) actions to minimize radiological release. For the SOARCA scenarios, successful implementation of this equipment and procedures would prevent core damage or delay or prevent radiological release; (6) improved phenomenological understanding of influential processes such as the following: in-vessel steam explosions, Mark I containment drywell (DW) shell attack, dominant chemical forms for fission products, direct containment heating, hot-leg creep rupture, reactor pressure vessel (RPV) failure and molten core-concrete interactions (MCCI); (7) and changes in plant operation that have occurred over time, including power uprates and higher core burnups

The SOARCA project, conducted by the NRC and Sandia National Laboratories (SNL), is a research effort to realistically estimate the outcomes of postulated severe accident scenarios that might cause a nuclear power plant (NPP) to release radioactive material into the environment. The SOARCA project applies many years of national and international reactor safety research, and incorporates the improvements and changes as noted above. The SOARCA project uses MELCOR (i.e., an integral severe accident analysis code) to model the severe accident scenarios within the plant, and MACCS2 (i.e., a radiological consequence assessment code) to model the offsite health consequences for any atmospheric releases of radioactive material. These models also consider onsite and offsite actions — including the implementation of mitigation measures and protective actions for the public such as evacuation and sheltering — that may prevent or mitigate accident consequences. For each scenario analyzed, SOARCA analyzed both a “mitigated” scenario assuming that the 10CFR50.54(hh) enhancements are successful, and an “unmitigated” scenario that does not credit the 10CFR50.54(hh) enhancements. Finally, the NRC has incorporated insights from health physics organizations and used both linear-no-threshold and alternate dose truncation models for analyzing health effects results. The SOARCA best estimate calculations, results, and conclusions are documented in NRC draft report NUREG-1935 [1], and NUREG/CR-7110, Volume 1, for the Peach Bottom pilot plant [2]. Figure 1 below shows a summary of the SOARCA individual latent cancer fatality (LCF) risk results, on a logarithmic scale. To provide perspective between SOARCA results and the more conservative estimates of severe reactor accident outcomes found in earlier NRC publications, SOARCA results are compared to the results of one of these previous publications: NUREG/CR-2239, “Technical Guidance for Siting Criteria Development,” commonly referred to as the 1982 Siting Study [3].

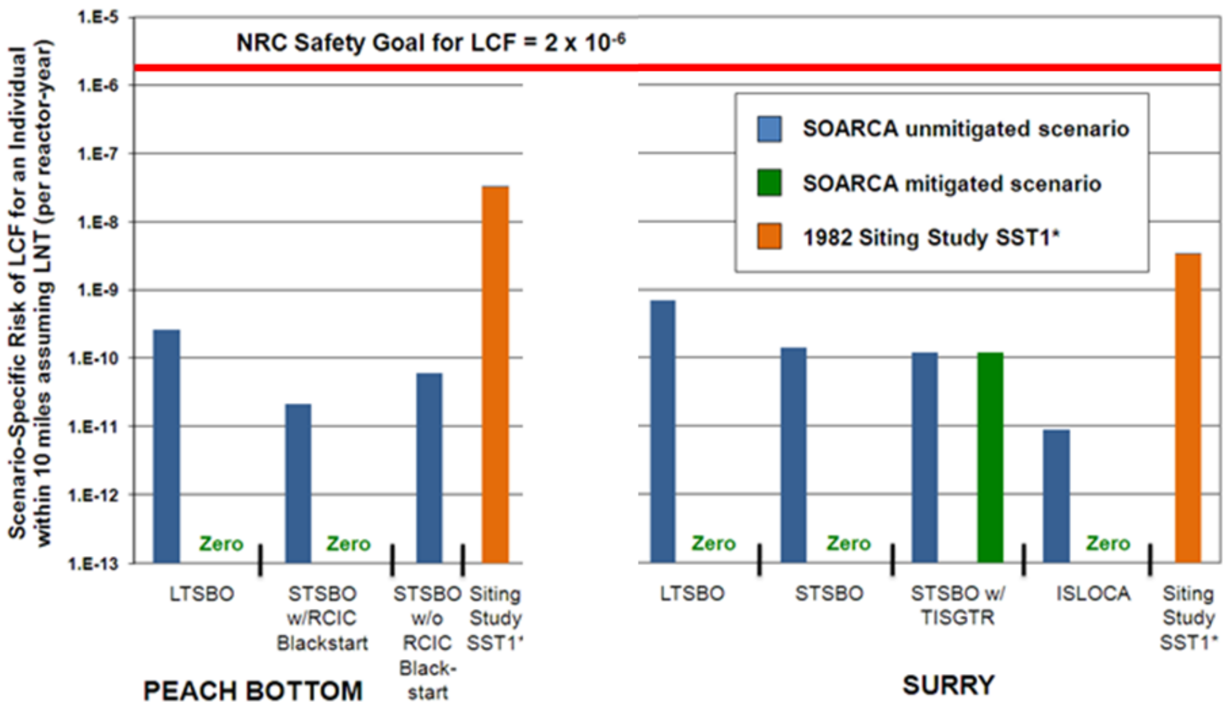

Figure 1. Comparison of individual LCF risk results for SOARCA mitigated and unmitigated scenarios to the NRC Safety Goal and to extrapolations of the 1982 Siting Study SST1

1.2. Objectives of the Uncertainty Study

The goals of this uncertainty analysis are to develop insight into the overall sensitivity of the SOARCA results to uncertainty in key modeling inputs, and identify the most influential input parameters for releases and consequences. Since this is a first-of-a-kind analysis in some ways, an additional goal is to demonstrate uncertainty analysis methodology that could be used in future source term, consequence and Level 3 PRA studies. This study leverages the existing SOARCA best estimate models and software, along with a representative set of key parameters.

The SOARCA best-estimate project included a number of sensitivity studies to examine issues associated with accident progression, mitigation, and offsite consequences for the accident scenarios of interest. The objective of these sensitivity studies was to examine specific issues and ensure the robustness of the conclusions documented in NUREG-1935. Single sensitivity studies, however, do not form a complete picture of the uncertainty associated with accident progression and offsite consequence modeling. Such a picture requires a more comprehensive and integrated evaluation of modeling uncertainties.

In general terms, the ‘best-estimate’ offsite consequence results presented in NUREG-1935 reflect only the uncertainty associated with weather conditions at the time of the accident scenario considered. These best-estimate offsite consequence values represent the expected (i.e., mean) value of the probability distribution obtained from a large number of aleatory weather trials. The impact of epistemic model parameter uncertainty is explored in detail for this uncertainty study by randomly sampling distributions for key epistemic model parameters that were considered to have a potential impact on the offsite consequences. Assessing key MELCOR and MACCS2 modeling uncertainties in an integrated fashion, yields an understanding of the relative importance of each uncertain input on the potential consequences. A detailed uncertainty study was performed for a single-accident scenario rather than all of the SOARCA sequences. The Peach Bottom Unmitigated LTSBO scenario was selected. While one scenario may not provide a complete picture with respect to all possible effects of uncertainties on both SOARCA pilot plants, it can be used as a first step to develop insight into the overall sensitivity of SOARCA results to input uncertainty.

2. APPROACH

2.1. Accident Scenario Selection

An accident sequence begins with the occurrence of an initiating event (e.g., a loss of offsite power, a loss-of-coolant accident (LOCA), or an earthquake) that perturbs the steady state operation of the NPP. The initiating event challenges the plant’s control and safety systems, whose failure could potentially cause damage to the reactor fuel and result in the release of radioactive fission products. Because a NPP has numerous diverse and redundant safety systems, many different accident sequences are possible depending on the type of initiating event that occurs, the amount of equipment that fails, and the nature of the operator actions involved, as described in the SOARCA study [1, 2].

The selection process for analysis of a SOARCA scenario for the uncertainty study considered both the magnitude of the radiological release and timing of the offsite release; a major impact on both early and latent cancer fatality risks. In this respect, an examination of the candidate SOARCA sequences considered timing, both the timing of core damage and the timing of containment failure. Station Blackout scenarios are important for BWRs. For Peach Bottom, the short-term station blackout (STSBO) frequency is roughly an order of magnitude lower than the LTSBO; however, the STSBO has a more prompt radiological release and a slightly larger release over the same interval of time. Although it was a more prompt release (i.e., 8 hours versus 20 hours), the STSBO release was delayed beyond the time needed for successful evacuation. The NUREG-1935 analysis indicated the absolute risk is indeed smaller for the STSBO than for the LTSBO (see Figure 1). The same trends apply for the Surry sequences where the lower-frequency sequences may have greater conditional risk but absolute risk is smaller than or equivalent to other higher-frequency sequences. Based on this rationale, the Peach Bottom Unmitigated LTSBO scenario was selected. The LTSBO release timing and consequences are characteristic of the majority of offsite consequences analyzed in SOARCA. Additionally, the performance of the safety relieve valve (SRV) as it impacts the main steam line (MSL) failure in the LTSBO scenario was an important sensitivity study identified by the peer review committee.

2.2. Selection of Uncertain Parameters

For the uncertainty study, a set of 23 epistemic MELCOR parameters, 20 independent MACCS2 epistemic parameters, and one MACCS2 aleatory parameter were selected. A discussion of the selected parameters is provided in Sections 4.1 and 4.2 for the MELCOR and MACCS2 parameters respectively. A core team of senior staff members from Sandia National Laboratories (SNL) and the NRC was formed with special expertise in probability and statistics, uncertainty analysis, MELCOR modeling, and MACCS2 consequence analysis. In addition, selected subject matter experts (SMEs) provided support on an as-needed basis and facilitated the reviews of data, parameters, and model abstractions. This approach is based on a formalized Phenomena Identification and Ranking Table (PIRT) process. The parameters and distributions selected were technically reviewed both internally and by the SOARCA Peer Review Committee.

2.3. Treatment of Uncertainty

In the design and implementation of analyses for complex systems, it is useful to distinguish between two types of uncertainty: aleatory uncertainty and epistemic uncertainty [4]. It is also important to note that some parameters may have both aleatory and epistemic attributes, but are treated as epistemic for analytic convenience. Aleatory uncertainty arises from an inherent randomness in the properties or behavior of the system under study. For example, the weather conditions at the time of a reactor accident are inherently random with respect to our ability to predict the future. Other potential examples include the variability in the properties of a population of system components and the variability in the possible future environmental conditions to which a system component could be exposed. Alternative designations for aleatory uncertainty include variability, stochastic, irreducible and type A. Epistemic uncertainty derives from a lack of knowledge about the appropriate value to use for a quantity that is assumed to have a fixed value in the context of a particular analysis. For example, the maximum temperature that a system can withstand before failing may not be known with certainty. Alternative designations for epistemic uncertainty include state of knowledge, subjective, reducible and type B.

The analysis of a complex system typically involves answering the following three questions about the system (the Kaplan and Garrick risk triplet [5]) and one additional question about the analysis itself:

1. What can happen?
2. How likely is it to happen?
3. What are the consequences if it happens?
4. How much confidence exists in the answers to the first three questions?

The answers to questions one and two involve the characterization of aleatory uncertainty, and the answer to question four involves the characterization and assessment of epistemic uncertainty, which is the objective of this study. The answer to question three typically involves numerical modeling of the system conditional on specific realizations of aleatory and epistemic uncertainty.

In the modeling system used to generate the SOARCA best estimate results, weather is treated as an aleatory parameter. Each best estimate calculation represents the mean offsite consequence for a given accident sequence calculated from a large number of weather trials. In this way, the SOARCA best estimate calculation seeks an answer to the question, "What is the expected consequence of a given accident scenario, like a LTSBO, at the Peach Bottom Atomic Power Station?" (i.e., expected outcome over all aleatory sequences) as opposed to, "What is the expected consequence of a given accident scenario during a snow storm in February at the Peach Bottom Atomic Power Station?" (i.e., results conditional on a specific weather trial). The SOARCA "best estimate" consequences including weather uncertainty is illustrated in Figure 2. A single source term release S_{BE} dependent upon the best estimate input, $x_{i, BE}$, was used as input to a consequence analysis dependent upon the best estimate input, $y_{i,BE}$. The result is a distribution of consequences conditional on the best estimate values (i.e., Question 3 above), over the weather variability (i.e., Question 1 and Question 2 above). The mean value, $||H||$, is the mean of the complementary cumulative distribution function (CCDF) and is the mean consequence over the weather variability. However, to address question four (i.e., "How much confidence exists in the answer to the first three questions?"), a series of analyses must be conducted that quantify the effects of epistemic uncertainty in the system over all possible weather conditions.

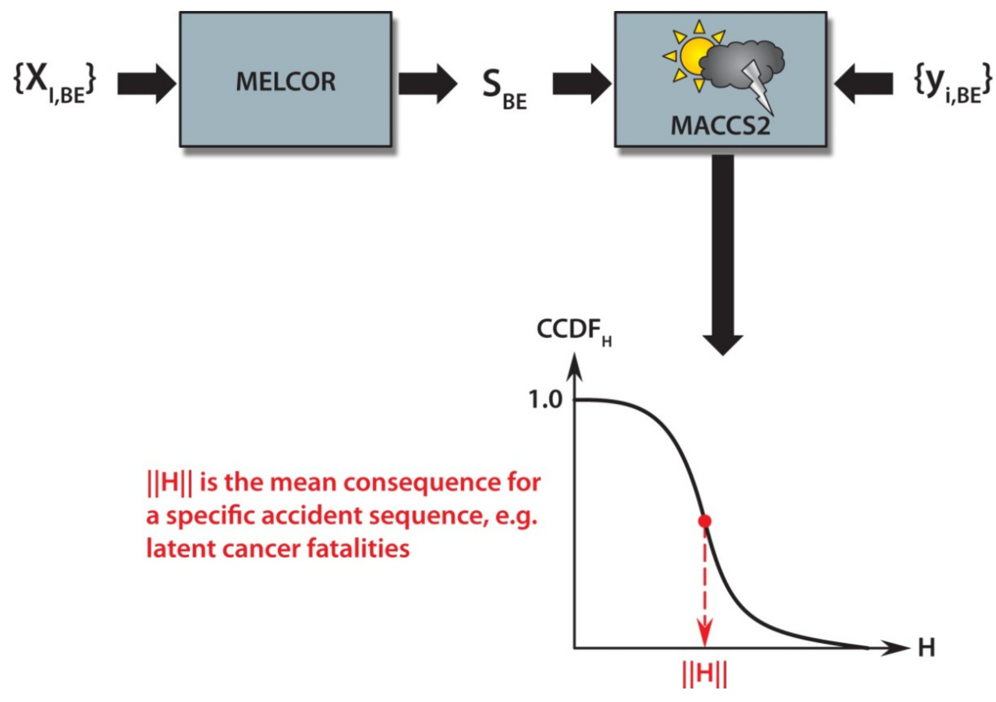

Figure 2. Typical complementary cumulative distribution function of consequence

3. DESCRIPTION OF ANALYSES

3.1. Software Used and Code Integration

MELCOR is a computer code that models the progression of severe accidents in PWRs and BWRs. MELMACCS compiles MELCOR outputs for transition into part of a MACCS (MELCOR Accident Consequence Code System) input. The MACCS Version 2 (MACCS2) can calculate the estimated consequences associated with a release of radioactive material into the environment. The following are the steps of this uncertainty analysis (Figure 3 shows the information flow): (1) Uncertain MELCOR and MACCS2 parameters are sampled; (2) MELCOR is run for each set of its N sampled values, generating N source terms; (3) MACCS2 is run for each set of its sampled values in conjunction with the associated MELCOR source term outputs.

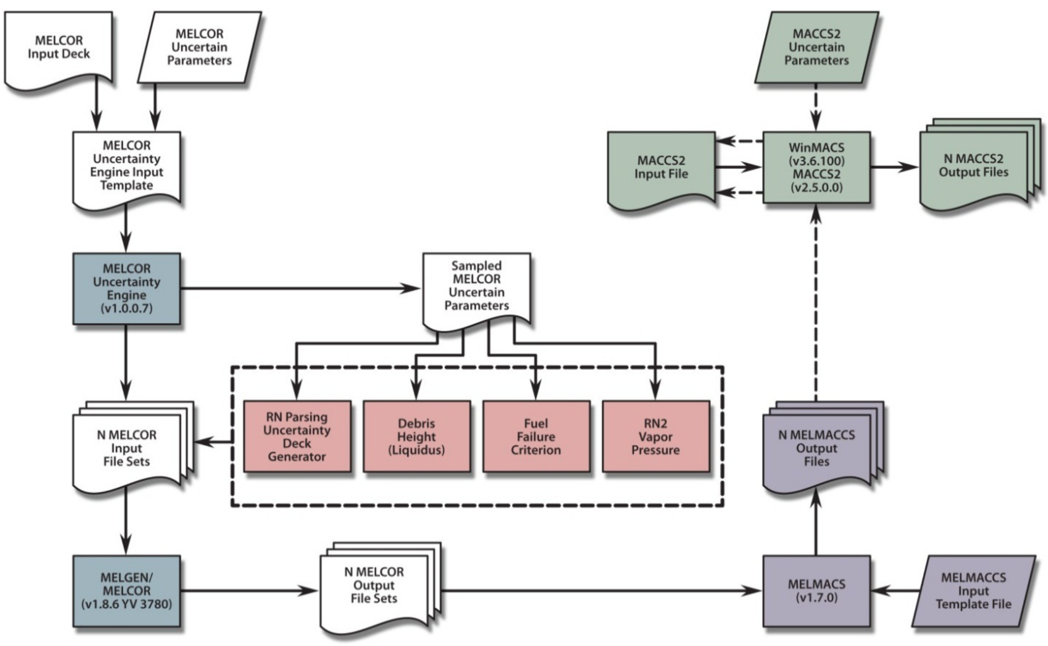

Figure 3. Conceptual representation of the information flow of the SOARCA uncertainty analysis

3.2. Uncertainty and Sensitivity Analysis

In the last step of a probabilistic approach results are statistically analyzed (via uncertainty analysis) and influence of input parameter uncertainty over the variance of the output is assessed (via sensitivity analysis). Many techniques have been developed to perform such analyses (several are presented in Helton et. al. [6]).

Formally, uncertainty analysis refers to the determination of the uncertainty in the analysis result that derives from the uncertainty in analysis inputs. This corresponds essentially to a statistical analysis of the set of output resulting from the epistemic sample set. Since separation of aleatory and epistemic uncertainty is considered, results can be presented in different ways. One way is to plot the output of interest as a set of CCDFs. Each CCDF (representing the effect of aleatory uncertainty) gives an answer to the Kaplan and Garrick risk triplet questions presented above. A steep curve will represent a low aleatory uncertainty (i.e., most of the values are close to each other) while a broad distribution of consequences will represent large aleatory uncertainty (i.e., the values vary significantly) (see Figure 4. A set of these CCDFs (representing the effect of epistemic uncertainty) represents our state of knowledge on this 'risk' (see Figure 5).

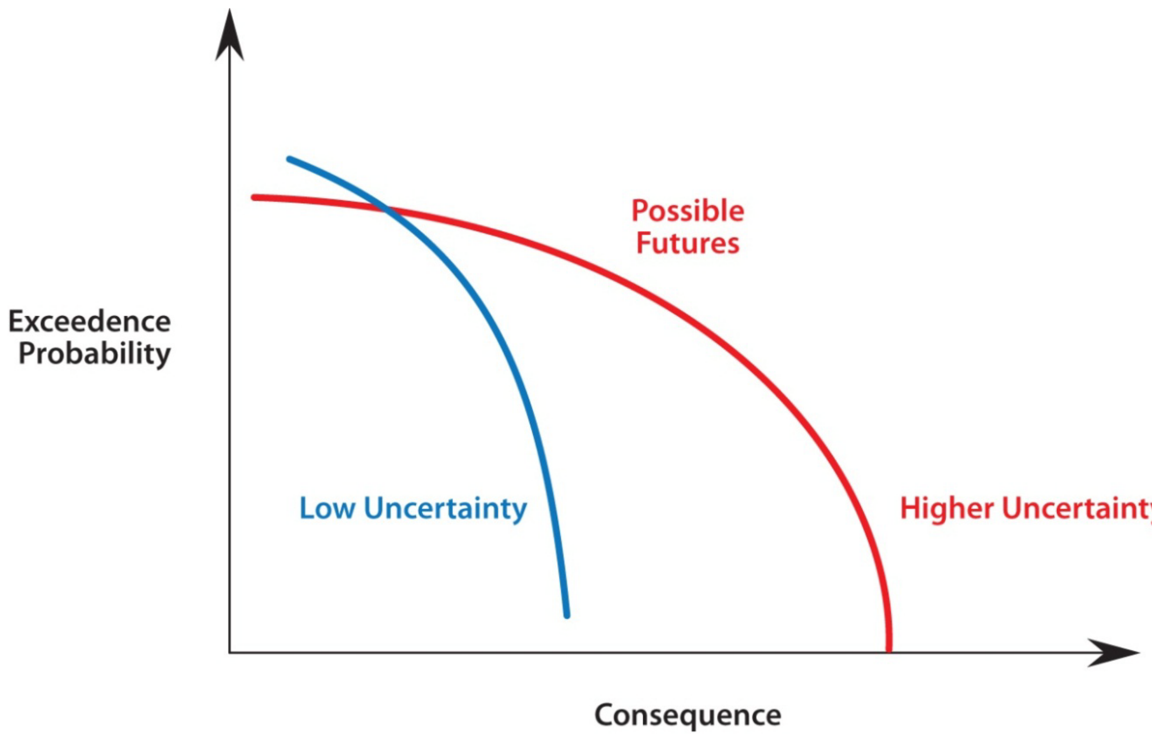

Figure 4. Consequence versus exceedence probability

If the CCDFs are close to each other, then the epistemic uncertainty is small (and reducing it will not result in much improvement). If they are spread out, then the epistemic uncertainty is large and one can gain accuracy if it is reduced. The display of these curves is typically used for the total life time risk of early or latent cancer fatality and helps understanding which uncertainty type is dominant -- aleatory (non-controlled) or epistemic (due to lack of knowledge).

Source term results from MELCOR, provides several estimates at each time step representing the (epistemic) uncertainty over the result of interest, due to lack of knowledge. This result can be represented as a function over time. The spread of the multiple time-dependent curves represents the effect of epistemic uncertainty, the same way the spread of CCDFs does. Often statistics such as the mean and quantiles (e.g., median and more extreme quantiles) are included in order to give a better visualization of this uncertainty. The quantiles will summarize the effect of epistemic uncertainty in input on the output of interest. When low probabilities are estimated, the accuracy of the estimate and its stability can be questioned. For assessing the stability of the estimate, confidence bounds can be calculated.

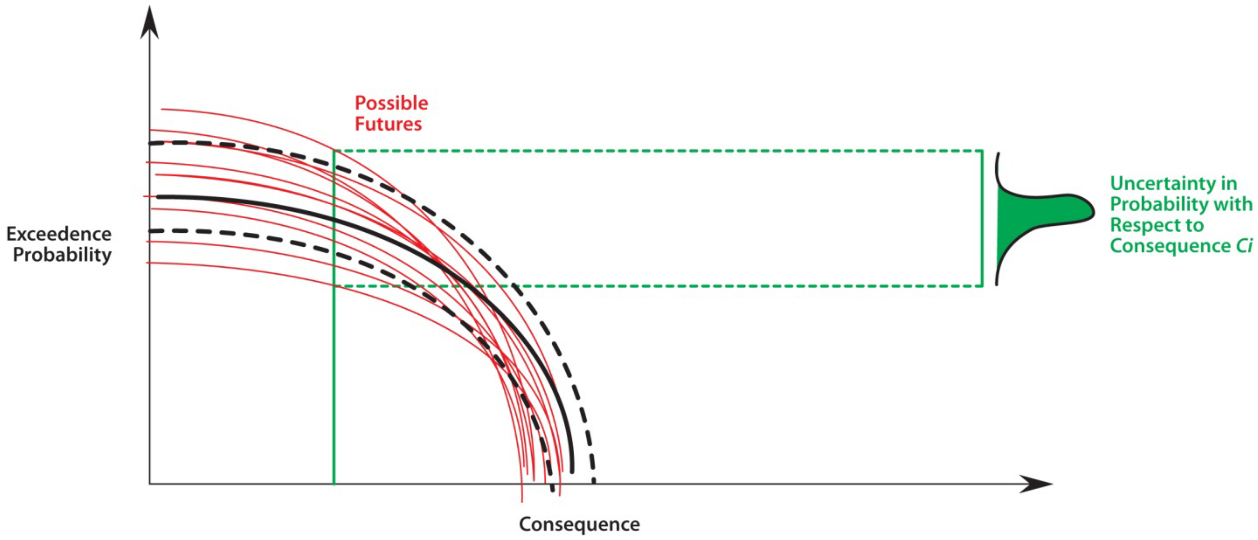

Figure 5. Uncertainty in exceedence probability with respect to consequence

Sensitivity analysis refers to the determination of the contributions of individual uncertain inputs to the uncertainty in results. To quantify and rank the importance of the uncertainty of each uncertain input on the variance of the output of interest, linear and rank regressions are used. Linear and rank regressions are fast and easy to implement and also give reliable estimates on the quality of their results. As rank regression often gives better results than linear regression, for essentially the same computing cost, Standardized Rank Regression Coefficients (SRRCs) are used in place of their parametric linear equivalent. The coefficient of determination of the regression model (R^2) will inform of the quality of the regression and consequently of the quality of the sensitivity analysis. A positive SRRC denotes a positive relation (in the sense that high values of input are associated with high values of output and low values of input are associated with low values of output) while a negative SRRC denotes a negative relation (for which high values of input are associated with low values of output and reciprocally). An approach that has shown to be efficient and the most appealing graphically in previous analyses, is to estimate the importance of some parameter in a stepwise fashion (i.e., using stepwise regression) for non-time-dependent parameters, and at a specific time step for time-dependent parameters.

In order to see the evolution of importance of input parameters over time, stepwise regression analyses are being applied at three different time steps for the source term results: 12 hours, 20 hours and 48 hours. For the offsite consequences, analysis of the mean and median are being analyzed.

Scatterplots of output versus input remains one of the simplest and most useful techniques and complete the classical stepwise regression in a more qualitative but graphically appealing way. When two parameters are leading the sensitivity analysis, instead of a classical scatterplot, 3D contour plots can be used, showing the conjoint influence of both input parameters.

Preliminary insights from the uncertainty analysis in progress indicate that there may be important threshold effects in the accident progression, and it may be important to look at the influence of combinations of two or more input parameters together [7]. Hence regression analyses on the partitioned sample space may also provide important insights. In addition, inspection of individual realizations of interest helps to both construct a phenomenological explanation of which uncertain inputs (or combinations of input and input ranges) are most influential on the results, as well as providing insights on potential threshold effects and input space partitions of interest.

4. UNCERTAIN INPUT PARAMETERS AND DISTRIBUTIONS

The uncertain parameters and their distributions were identified and characterized through an informal elicitation of subject matter experts, based on a formal PIRT process. The subject matter experts were asked to define distributions for the parameters which they considered most important in describing the uncertainty around the SOARCA best estimate analysis. In addition, the uncertain parameters and distributions were presented to and evaluated by the independent SOARCA Peer Review Committee, who provided comments

on the selected parameters and distributions. The parameters and distributions were revised to address these comments.

The general approach taken in defining the scope of the SOARCA uncertainty quantification is to attain a balanced depth and breadth of coverage of contribution from uncertainty across the spectrum of phenomena operative in the analyses without excessive detail dedicated to any particular regime of phenomenon. Both MELCOR and MACCS permit extensive access to parameters that may be uncertain, but for practical reasons, a judiciously selected subset of possible uncertain parameters is proposed that cover the range of phenomena across the stages of a severe accident and subsequent consequences. Table 1 shows both the MELCOR and MACCS2 parameter groups selected for this uncertainty analysis.

4.1. Source Term Uncertainty (MELCOR Inputs)

The MELCOR uncertain parameters are selected to cover the following issues and phenomenological areas:

- sequence issues
- in-vessel accident progression issues
- ex-vessel accident progression issues
- containment behavior issues
- fission product release, transport, and deposition

These broad areas span the temporal domain of the severe accident progression ranging from minor sequence variations as affected by safety relief valve (SRV) behavior, uncertainties in the core damage and melt progressions, especially those affecting rate of core degradation and amount of hydrogen generation. Hydrogen production provides an indication of fission product release from the fuel, since hydrogen generation is an indicator of cladding oxidation which is an indicator that fuel temperatures are rising above 1500 K (2240 F) on their way to 2200 K (3500 F) and above. This is the temperature range where thermally driven release of the volatile fission products occurs. Source term release behavior in terms of the rate and total amount released in-vessel is strongly coupled to in-vessel melt progression behavior owing to the strong temperature dependence of fission product release. The onset of volatile fission product release is set by the time of fuel heatup above about 1500K (2240 F), and this is tightly coupled to cladding oxidation rate. Total release of both volatile and lower volatile species is affected by the time at which fuel remains at elevated temperatures and the state of the fuel (rods or debris). Therefore, many of the parameters that affect cladding oxidation and hydrogen generation also strongly affect fission product release. Other parameters more specific to fission product transport include deposition processes (e.g., chemisorption or hygroscopicity) and settling processes (agglomeration shape factors for example). Speciation of Cesium and Iodine affects the precise volatility of Cesium and consequently affects both release and revaporization. The parameters selected in the study were considered in terms of both melt progression and fission product release and transport. Including, important behaviors taking place following vessel lower head melt-through such as melt attack of the drywell (DW) liner, containment behavior issues, such as uncertainty in onset of DW head flange leakage, and uncertainties in radioactive aerosol transport mechanics. The selection of uncertain parameters ensures representation of uncertainties in the major phases of the accident evolution.

4.2. Consequence Uncertainty (MACCS2 Inputs)

The MACCS2 consequence model is used in the SOARCA analysis to calculate offsite doses and their effect on members of the public. The epistemic uncertainty was considered for the principal phenomena in MACCS2, including atmospheric transport using a straight-line Gaussian plume model of short-term and long-term dose accumulation through several pathways including: cloudshine, groundshine, and inhalation. The ingestion pathway was not treated in the SOARCA analyses because uncontaminated food and water supplies are abundant within the United States and it is unlikely that the public would eat radioactively contaminated food. The parameter uncertainty in the MACCS2 consequence model will impact the following doses included in the SOARCA best estimate reported risk metrics: cloudshine during plume passage, groundshine during the emergency and long-term phases from deposited aerosols, inhalation during plume passage and following plume passage from resuspension of deposited aerosols. Resuspension is treated during both the emergency and long-term phases.

Development of the emergency planning related uncertainty parameters for MACCS2 input required establishing an emergency response timeline. The timeline includes actions described in the onsite and offsite emergency response plans. The emergency response plans are tested and exercised often and there is a high confidence in the interactions between onsite and offsite agencies. Research of existing evacuations provided information regarding movement of the public in response to an emergency and has shown that emergency response actions are routinely implemented and successful (see discussion and references in [1,2]). Although there is high confidence in response actions, an emergency response is a dynamic event with uncertainties in elements of the response.

Table 1. SOARCA uncertainty analysis parameter groups

MELCOR	**MACCS2**
Epistemic Uncertainty	**Epistemic Uncertainty**
Sequence Issues	*Deposition*
SRV stochastic failure to reclose	Wet deposition model
Battery Duration	Dry deposition velocities
In-Vessel Accident Progression Parameters	*Shielding Factors*
Zircaloy melt breakout temperature	Shielding factors
Molten clad drainage rate	*Early Health Effects*
SRV thermal seizure criterion	Early health effects (EFFACA, EFFACB, EFFTHR)
SRV open area fraction	*Latent health effects*
Steam line creep rupture area	Groundshine
Fuel failure criterion	Dose and dose rate effectiveness factor
Radial debris relocation time constants	Mortality risk coefficient
Ex-Vessel Accident Progression Parameters	Inhalation dose coefficients
Debris lateral relocation – cavity spillover and spreading rate	*Dispersion Parameters*
Containment Behavior Parameters	Linear, crosswind dispersion coefficients
Drywell liner failure flow area	Linear, vertical dispersion coefficients
Hydrogen ignition criteria	*Relocation Parameters*
Railroad door open fraction	Hotspot relocation
Drywell head flange leakage (K, E, δ)	Normal relocation
Chemical Forms of Iodine and Cesium	*Evacuation Parameters*
Iodine and Cesium fraction	Evacuation delay
Aerosol Deposition	Evacuation speed
Dynamic and agglomeration shape factors	**Aleatory Uncertainty**
Particle Density	Weather Trials

All of the emergency planning parameters used in MACCS2 were reviewed to determine the most appropriate parameters for the uncertainty analysis. The following three[1] emergency planning parameter sets were selected: hotspot and normal relocation, evacuation delay, and evacuation speed.

5. CONCLUSION

The overall objective of the SOARCA project is to develop a body of knowledge on the realistic outcomes of severe reactor accidents using self-consistent, integrated modeling of accident progression and offsite consequences drawn from current best practices modeling to estimate offsite consequences for important classes of events. This paper reports on the SOARCA uncertainty analysis that is in progress for the Peach Bottom boiling-water reactor pilot plant and "unmitigated" long-term station black-out scenario. This uncertainty analysis leverages the existing SOARCA best estimate models and software. The uncertain parameters and their distributions were identified and characterized through an informal elicitation of subject matter experts. The uncertainty is propagated to consequences results using a two-step Monte Carlo process. Standard regression-based techniques, investigation of individual Monte Carlo realizations of interest, and the search for partitions of interest in the input parameter sample space are being used to identify the most influential parameters, combinations of parameter ranges, and corresponding phenomenology.

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References

[1] U.S. Nuclear Regulatory Commission, NUREG-1935, "State-of-the-Art Reactor Consequence Analyses (SOARCA) Report," Washington, DC, 2012.

[2] Sandia National Laboratories, NUREG/CR-7110, "State-of-the-Art Reactor Consequence Analyses Project Volume 1: Peach Bottom Integrated Analysis," U.S. Nuclear Regulatory Commission, Washington, DC, 2012.

[3] U.S. Nuclear Regulatory Commission, NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," Sandia National Laboratories, Albuquerque, NM, 1982

[4] Helton, J.C. and D.E. Burmaster, "Guest editorial: Treatment of Aleatory and Epistemic Uncertainty in Performance Assessments for Complex Systems," *Reliability Engineering & System Safety*, **54**(2-3): p. 91-94, 1996.

[5] Kaplan, S. and B.J. Garrick, "On The Quantitative Definition of Risk," *Risk Analysis*, **1**(1): p. 11-27, 1981.

[6] Helton, J.C., et al., "Survey of Sampling-based Methods for Uncertainty and Sensitivity Analysis," *Reliability Engineering & System Safety*, **91**(10-11): p. 1175-1209, 2006.

[7] Ghosh, S.T. and G.E. Apostolakis, "Extracting Risk Insights from Performance Assessments for High-Level Radioactive Waste Repositories," *Nuclear Technology,* **153**: p. 70-88, 2006.

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[1] The habitability criterion is also considered to be an important potentially uncertain parameter, but will not be included as part of the integrated uncertainty analysis. A separate discussion will be included in the final SOARCA uncertainty analysis report to discuss insights on the influence of the habitability criterion based on past sensitivity studies and uncertainty analysis demonstration cases.