SOARCA Peach Bottom Atomic Power Station Long-Term Station Blackout Uncertainty Analysis: Overview

S. Tina Ghosh[a*], Patrick D. Mattie[b], Randall O. Gauntt[b], Nathan E. Bixler[b], Kyle W. Ross[b], Cedric J. Sallaberry[b], and Douglas M. Osborn[b]

[a]Nuclear Regulatory Commission, Washington, DC, USA

[b]Sandia National Laboratories, Albuquerque, NM, USA

Abstract: This paper provides an overview of the uncertainty analysis for the accident progression, radiological releases, and offsite consequences for the State-of-the-Art Reactor Consequence Analyses (SOARCA) unmitigated long-term station blackout accident scenario at the Peach Bottom Atomic Power Station. The SOARCA project (NUREG-1935) estimated the outcomes of postulated severe accident scenarios which could result in release of radioactive material from a nuclear power plant into the environment. The SOARCA model was based on best practices used to estimate offsite consequences of important classes of events. SOARCA coupled the deterministic 'best estimate' modeling of accident progression (i.e., reactor and containment thermal-hydraulic response and fission product transport), embodied in the MELCOR code with modeling of offsite consequences in MACCS2. This uncertainty analysis presents the results of an integrated analysis of epistemic parameter uncertainty associated with the accident progression and offsite consequence modeling. This uncertainty analysis supported the overall conclusions of the SOARCA project and provided some new insights.

Keywords: SOARCA, MACCS, MELCOR, Uncertainty Analysis, Severe Accident Analysis

1. INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC), the nuclear power industry, and the international nuclear energy research community have devoted considerable research over the last several decades to examining severe reactor accident phenomena and offsite consequences. The NRC initiated the State-of-the-Art Reactor Consequence Analyses (SOARCA) project [1] to leverage this research and develop best estimates of the offsite radiological health consequences for potential severe reactor accidents for two pilot plants: the Peach Bottom Atomic Power Station in Pennsylvania and the Surry Power Station in Virginia. By applying modern analysis tools and techniques, the SOARCA project developed a body of knowledge regarding the realistic outcomes of select severe nuclear reactor accidents. To accomplish this objective, the SOARCA project's integrated modeling of accident progression and offsite consequences used both state-of-the-art computational analysis tools (the MELCOR code and the MELCOR Accident Consequence Code System, Version 2 [MACCS2]) and best modeling practices drawn from the collective wisdom of the severe accident analysis community. The SOARCA project is documented in NUREG-1935 (2012) [1], NUREG/CR-7110, Volume 1, and NUREG/CR-7110 Volume 2 (2013) [2] [3].

This paper provides an overview of the NRC's uncertainty analysis of the SOARCA unmitigated long-term station blackout (LTSBO) severe accident scenario for the Peach Bottom Atomic Power Station [4]. The objectives of this uncertainty analysis are to evaluate the robustness of the SOARCA project's deterministic results and conclusions, and to develop insight into the overall sensitivity of the SOARCA results to uncertainty in key modeling inputs. As this is a first-of-a-kind analysis in its integrated look at uncertainties in MELCOR accident progression and MACCS2 offsite consequence analyses, an additional objective is to demonstrate uncertainty analysis methodology that could be used in future combined Level 2/3 probabilistic risk assessment (PRA) studies.

* tina.ghosh@nrc.gov

2. SCOPE AND APPROACH

The SOARCA offsite consequence results presented in NUREG-1935 incorporated only the aleatory uncertainty associated with weather conditions at the time of the accident. The reported offsite consequence values represent the average (mean) value obtained from a large number of aleatory weather trials. The weather uncertainty is handled the same way in this uncertainty analysis. In addition, the impact of state-of-knowledge (epistemic) uncertainty in the input parameters is explored in detail by randomly sampling distributions for model parameters that are considered to be potentially important. Assessing key MELCOR and MACCS2 parameter uncertainties in an integrated fashion helps form an understanding of the relative importance of each uncertain input on the potential consequences.

This analysis uses the existing SOARCA models and software. In other words, the uncertainty stemming from the choice of conceptual models and model implementation is not explicitly explored, nor is completeness uncertainty. It is worth noting, however, that many of the input parameters in the SOARCA models are lumped parameters that can represent different mechanistic models. In that respect, the distributions assigned to some input parameters can serve as a substitute for exploring mechanistic model uncertainty as well. In addition, not all possible uncertain input parameters were included in the analysis. Rather, a set of key parameters was carefully chosen to capture important influences on the potential release of radioactive material to the environment and on offsite health consequences.

A detailed uncertainty analysis was performed for one accident scenario rather than all seven of the SOARCA scenarios documented in NUREG-1935. The SOARCA Peach Bottom boiling water reactor (BWR) Pilot Plant unmitigated, LTSBO scenario is analyzed. While one scenario cannot provide a complete exploration of all possible effects of uncertainties in analyses for the two SOARCA pilot plants, it can be used to provide initial insights into the overall sensitivity of SOARCA results and conclusions to input uncertainty. In addition, since station blackouts are an important class of events for BWRs in general, the phenomenological insights gained on accident progression and radionuclide releases may prove useful for BWRs in general.

Through informal expert elicitation and iteration after interim reviews by the independent SOARCA peer review panel and the NRC's Advisory Committee on Reactor Safeguards, 21 key MELCOR parameters and 20 key MACCS2 parameter groups (see Table 1) were identified for inclusion in the uncertainty analysis, and distributions were defined for each parameter. The 20 MACCS2 parameter groups are comprised of 350 individual parameters, many of which are fully correlated and form a parameter group. For example, there are many individual organ-specific and radionuclide-specific dose conversion factors, which are considered one group of parameters.

The MELCOR uncertain parameters were selected to capture:

- accident sequence issues,
- accident progression issues within the reactor vessel,
- accident progression issues outside the reactor vessel,
- containment behavior issues, and
- fission product release, transport, and deposition within plant structures.

These broad areas span the severe accident progression over time, ranging from minor sequence variations as affected by safety relief valve (SRV) behavior, to uncertainties in the core damage and melt progressions. Other parameters more specific to fission product transport include deposition and settling processes, and chemical speciation of cesium and iodine which affects both release and transport within plant structures.

Table 1. SOARCA Peach Bottom Analysis Uncertain Parameter Groups

MELCOR	MACCS2
Epistemic Uncertainty	**Epistemic Uncertainty**
Sequence Issues	***Deposition***
SRV stochastic failure to reclose (SRVLAM)	Wet deposition model (CWASH1)
Battery Duration (BATTDUR)	Dry deposition velocities (VEDPOS)
In-Vessel Accident Progression Parameters	***Shielding Factors***
Zircaloy melt breakout temperature (SC1131(2))	Shielding factors (CSFACT, GSFAC, PROTIN)
Molten clad drainage rate (SC141(2))	***Early Health Effects***
SRV thermal seizure criterion (SRVFAILT)	Early health effects (EFFACA, EFFACB, EFFTHR)
SRV open area fraction (SRVOAFRAC)	***Latent health effects***
Main Steam line creep rupture area fraction (SLCRFRAC)	Groundshine (GSHFAC)
Fuel failure criterion (FFC)	Dose and dose rate effectiveness factor (DDREFA)
Radial debris relocation time constants (RDMTC, RDSTC)	Mortality risk coefficient (CFRISK)
Ex-Vessel Accident Progression Parameters	Inhalation dose coefficients (radionuclide specific)
Debris lateral relocation – cavity spillover and spreading rate (DHEADSOL, DHEADLIQ)	***Dispersion Parameters***
Containment Behavior Parameters	Crosswind dispersion coefficients (CYSIGA)
Drywell liner failure flow area (FL904A)	Vertical dispersion coefficients (CZSIGA)
Hydrogen ignition criteria (H2IGNC)	***Relocation Parameters***
Railroad door open fraction (RRIDRFAC, RRODRFAC)	Hotspot relocation (DOSHOT, TIMHOT)
Drywell head flange leakage (K, E, δ)	Normal relocation (DOSNRM, TIMNRM)
Chemical Forms of Iodine and Cesium	***Evacuation Parameters***
Iodine and Cesium fraction (CHEMFORM)	Evacuation delay (DLTEVA)
Aerosol Deposition	Evacuation speed (ESPEED)
Particle Density (RHONOM)	**Aleatory Uncertainty**
	Weather Trials

The parameters selected from the MACCS2 consequence model are those that affect individual latent cancer fatality (LCF) risk and individual early fatality risk, due to:

- cloudshine during contaminant plume passage,
- groundshine from deposited contaminants, and
- inhalation during plume passage and following plume passage from resuspension of deposited contaminants.

Parameters related to emergency planning were also varied. Although there is high confidence in emergency response actions, an emergency is a dynamic event with uncertainties in elements of the response. The following three emergency planning parameter sets were selected:

- hotspot and normal relocation,
- evacuation delay, and
- evacuation speed.

Several of the distributions for non-site-specific MACCS2 parameters selected for this analysis are based on expert elicitation data. The United States and the Commission of European Communities conducted a series of expert elicitations in the 1990's to obtain distributions for uncertain variables used in health consequence analyses related to accidental release of nuclear material. The distributions reflect degrees of belief for non-site specific parameters that are uncertain and are likely to have significant or moderate influence on the results [5].

This uncertainty analysis uses a two-step Monte Carlo simulation. Simple random sampling was chosen for MELCOR calculations as some of the results do not converge. 865 of the 900[1] MELCOR realizations ran to completion. From these complete MELCOR realizations, a family of source term results was produced. Latin hypercube sampling was chosen for WinMACCS, with a sample size of 865 to match the number of source terms. The MACCS2 results are presented as individual LCF risk and individual early fatality risk, averaged over the aleatory uncertainty stemming from weather.

Four regression techniques are used in this analysis to estimate the importance of the input parameters with respect to the uncertainty in source terms and consequences: linear rank regression, quadratic regression, recursive partitioning, and multivariate adaptive regression splines (MARS). This analysis provides measures of the effects of the selected uncertain parameters both individually and in interaction with other parameters (the T_i index in the three more advanced methods), and helps:

- Identify which uncertainty in these important parameters and phenomena are driving the variability in model results.
- Verify and validate the SOARCA model through exploration of unexpected or non-physical phenomena in the distributions of results.
- Provide an assessment of the regression techniques and uncertainty analysis approach.
- Provide a basis for future work.

3. SOURCE TERM RESULTS AND INSIGHTS

Performing the source term calculations of the Peach Bottom unmitigated LTSBO uncertainty analysis revealed three accident progression sub-scenarios within the Peach Bottom unmitigated LTSBO scenario: (1) early stochastic failure of the cycling SRV, which was the SOARCA best-estimate scenario in NUREG-1935; (2) thermal failure of the SRV without main steam line (MSL) creep rupture; and (3) thermal failure of the SRV with MSL creep rupture. The three sub-scenarios exhibit differences in releases, with MSL rupture generally leading to the largest releases since the benefit of wetwell scrubbing is lost. The majority of samples in this uncertainty analysis resulted in larger iodine and cesium releases than the SOARCA project calculated because early stochastic failure of the cycling SRV (the SOARCA best-estimate scenario) generally leads to smaller releases. Which of these accident progression sub-scenarios are realized depends on the values sampled for a couple of key uncertain variables: the SRV stochastic failure rate and the SRV open area fraction if the SRV fails thermally instead. The data supporting the distributions of these two variables is sparse to non-existent.

Similarly, there is considerable uncertainty in assessing what the distributions should be for other important variables, such as the size of the opening that results from core melt contacting and failing the drywell liner. This uncertainty analysis is most useful in uncovering important influences, and defining the plausible range in accident progression and consequences given uncertainty in the input parameters studied. The relative likelihood of different results within the range, on the other hand, still retains considerable uncertainty given the scarcity of relevant data to support the definition of some key input distributions.

Figure 1 shows the fraction of iodine core inventory released over time, for the 865 samples. For contrast, note the SST1 source term from the 1982 Siting Study (NUREG/CR-2239 [6]) assumed an

[1] The other 35 samples that did not run to completion were due to numerical convergence challenges, and not because of any problems with extending the MELCOR model to a larger parameter value domain.

iodine release starting at 1.5 hours, and steadily rising to a final value of 0.45 by 3 hours (as noted in NUREG-1935 for the SOARCA project [1]). The earliest releases in this uncertainty analysis begin after 10 hours, with average (mean) and 95th percentile iodine releases a factor of 10 and 4 smaller, respectively.

Figure 1. Time-Dependent Fraction of Iodine Core Inventory Released to the Environment for the First 48 Hours for Combined (865) Results for the Peach Bottom Unmitigated LTSBO

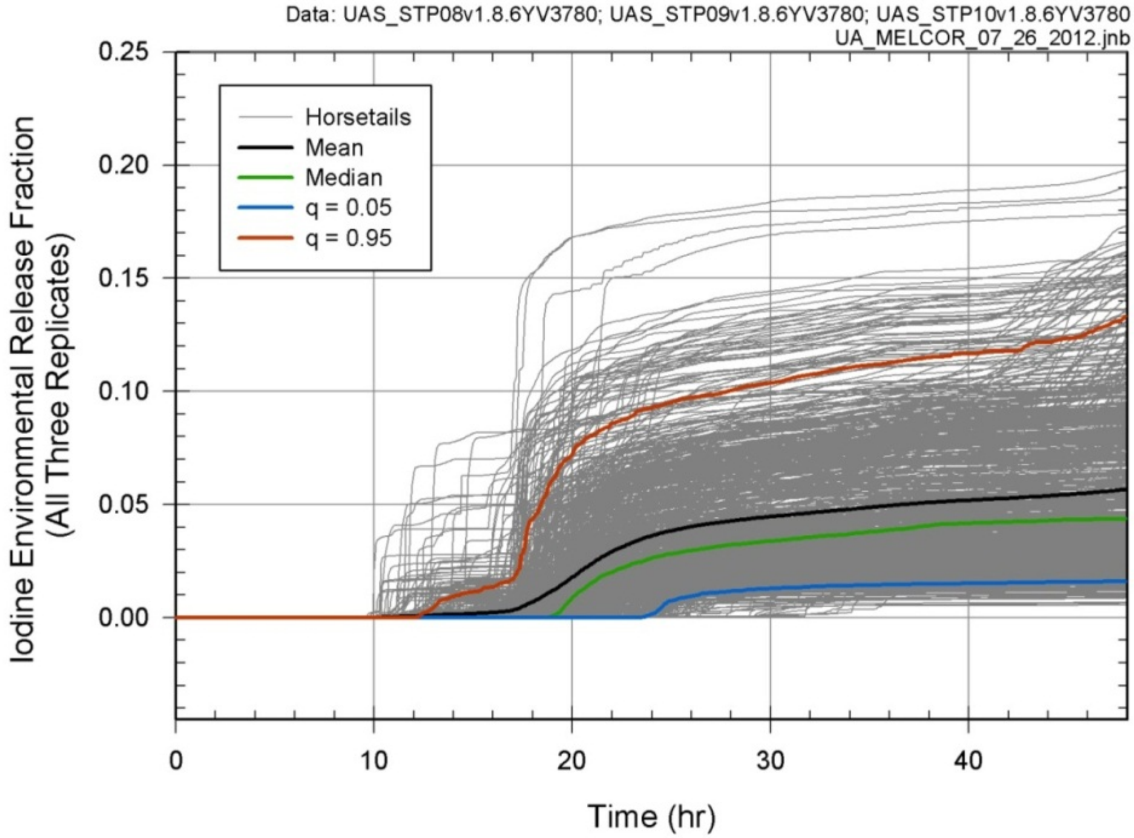


Figure 2 shows the fraction of cesium core inventory released over time, for the 865 samples. For contrast, the SST1 source term from the 1982 Siting Study assumed a cesium release starting at 1.5 hours and steadily rising to a final value of 0.67 by 3 hours. The earliest releases in this uncertainty analysis begin after 10 hours, with average (mean) and 95th percentile cesium releases a factor of 30 and 7 smaller, respectively.

Several influences were found to strongly affect the magnitude and timing of fission product releases to the environment, as summarized below. Most notably, with respect to the magnitude of the source term (the magnitude of cesium and/or iodine releases), the following were found to be influential:

- whether the sticking open of an SRV occurs before or after the onset of core damage, with higher releases if after core damage, and the SRV open area if the SRV fails thermally rather than stochastically,
- whether a MSL creep rupture occurs (largely determined by the two SRV factors above); if MSL rupture occurs, releases will be higher due to fission products being vented straight to the drywell and bypassing wetwell scrubbing,

- the amount of cesium chemisorbed (if any) from cesium hydroxide (CsOH) into the stainless steel of reactor pressure vessel (RPV) internals; more chemisorption results in less cesium release to the environment in high-temperature scenarios such as MSL rupture,
- whether core debris relocates from the RPV to the reactor cavity all at once or over an extended period of time with relocation all at once leading to lower releases to the environment,
- the degree of oxidation, primarily fuel-cladding oxidation, occurring within the vessel with greater oxidation resulting in larger releases, and
- whether a surge of water from the wetwell up onto the drywell floor occurs at drywell liner melt-through (which depends on the sampled value of the drywell liner open area), with the development of a surge leading to larger releases.

Figure 2. Time-Dependent Fraction of Cesium Core Inventory Released to the Environment for the First 48 Hours for Combined (865) Results for the Peach Bottom Unmitigated LTSBO

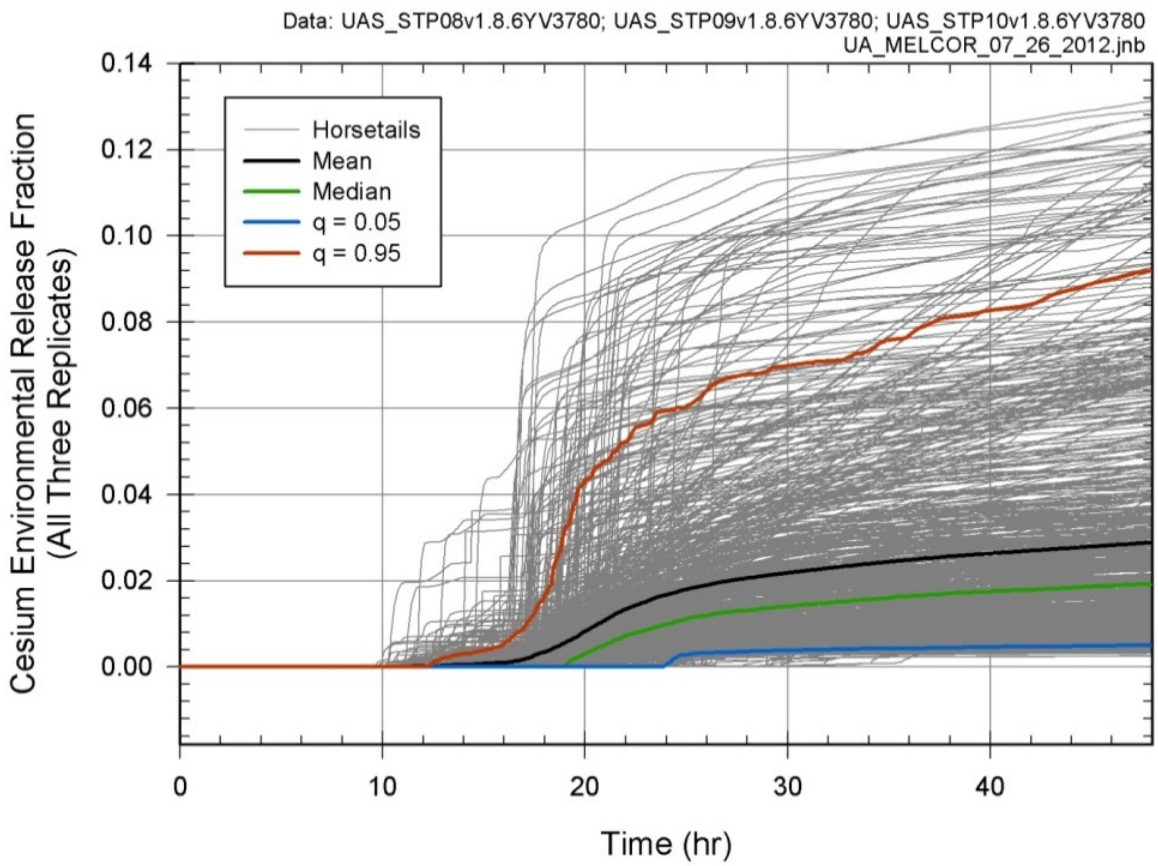


Table 2 shows regression results for the most important parameters with respect to magnitude of cesium release. The sampled parameters shown to strongly or meaningfully influence the magnitude of the fission product releases, because they contribute to the important phenomena noted above, were:

- The expected number of cycles an SRV can undergo before failing to reclose, SRVLAM (stochastic failure rate of SRV),
- The chemical form of cesium, CHEMFORM (i.e., the amount of cesium as CsOH opposed to Cs_2MoO_4),

- The size of the breach in the drywell liner from core debris contacting and melting through the liner, FL904A,
- The fractional open area of an SRV after it has failed to reseat because of overheating, SRVOAFRAC,
- The time-at-temperature criterion specified for loss of "intact" fuel rod geometry, FFC, and
- The temperature at which oxidized fuel cladding mechanically fails, SC1131_2.

Table 2. Regression Analyses of Fraction of Cesium Released Over 48 Hours

	Rank Regression			Quadratic			Recursive Partitioning			MARS		
Final R^2	**0.61**			**0.64**			**0.90**			**0.66**		
Input name	**R^2 inc.**	**R^2 cont.**	**SRRC**	**S_i**	**T_i**	**p-val**	**S_i**	**T_i**	**p-val**	**S_i**	**T_i**	**p-val**
SRVLAM	0.50	0.50	-0.72	0.39	0.64	0.00	0.43	0.70	0.00	0.57	0.68	0.00
FL904A	0.53	0.03	0.19	0.01	0.04	0.12	0.06	0.02	0.44	0.00	0.03	0.10
FFC	0.55	0.02	0.19	0.04	0.05	0.31	0.02	0.10	0.00	0.01	0.08	0.00
RRDOOR	0.58	0.03	0.33	0.02	0.10	0.00	0.02	0.03	0.19	---	---	---
SRVOAFRAC	0.59	0.02	-0.13	0.07	0.19	0.00	0.11	0.33	0.00	0.12	0.27	0.00
CHEMFORM	0.60	0.01	0.09	0.00	0.08	0.38	0.01	0.18	0.00	0.02	0.00	0.87
SC1131_2	0.60	0.01	-0.07	0.02	0.01	0.63	0.00	0.07	0.00	0.00	0.04	0.01
RRIDFRAC	0.61	0.00	0.06	0.05	0.00	1.00	0.00	0.01	0.57	0.01	0.03	0.07

With respect to release timing, the strongest influences identified were:

- when the reactor core isolation cooling (RCIC) system fails (determined solely by the time taken to exhaust the station batteries),
- when the SRV fails to reseat, and
- what the open fraction of the SRV is when it fails to reseat if it fails thermally.

4. OFF-SITE CONSEQUENCE RESULTS AND INSIGHTS

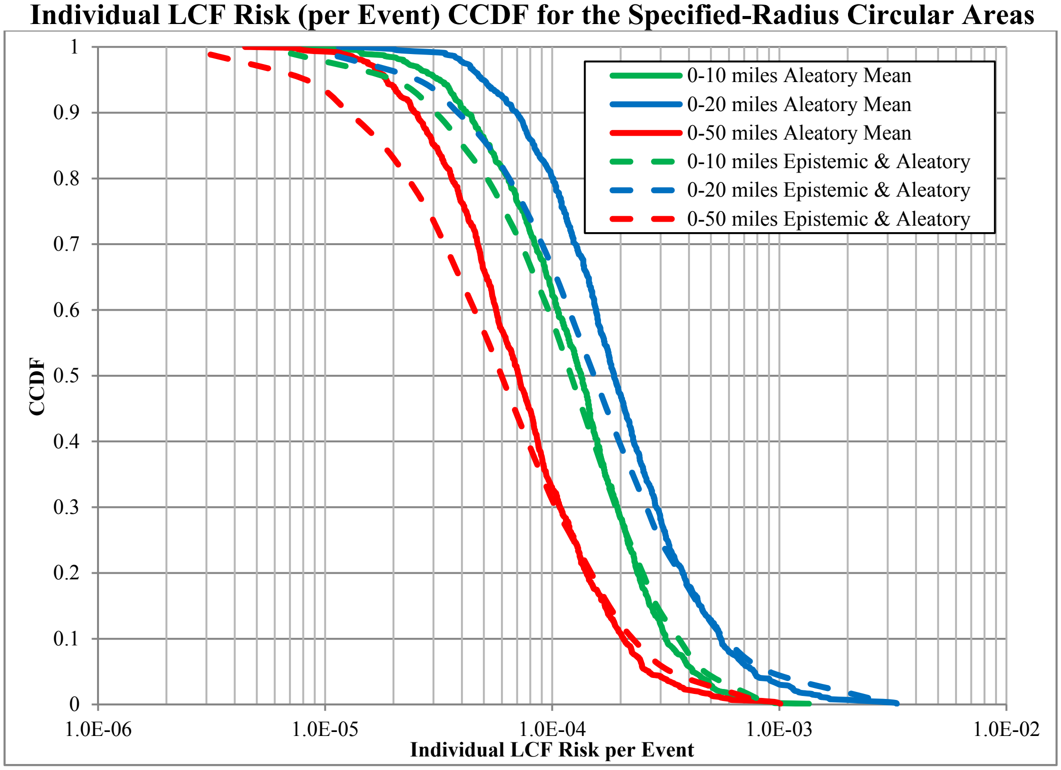
Table 3 shows the distribution of results for the conditional, mean (over weather), individual LCF risk within the 10-mile radius and 50-mile radius circular areas, using the linear-no-threshold dose-response model. For contrast, the 10-mile LCF risk recalculated for the SST1 source term in NUREG-1935 [1] is more than an order of magnitude higher than the 95th percentile from this uncertainty analysis.

Figure 3 shows the LCF risk complementary cumulative distribution functions (CCDFs) for the combined aleatory/weather and epistemic uncertainty (in dashed lines), and the epistemic uncertainty alone (solid lines), for 10-, 20-, and 50-mile radial areas around the plant.

Table 3. Conditional[2], Mean (Over Weather), Individual LCF Risk (Per Event) Statistics for the Uncertainty Analysis for 10-Mile Radius And 50-Mile Radius Circular Areas Around the Plant, Using All 865 MELCOR/MACCS2 Samples, and the Linear-No-Threshold Dose-Response Model

	0-10 miles	0-50 miles
5th percentile	3×10^{-5}	2×10^{-5}
Median	1×10^{-4}	7×10^{-5}
Mean	2×10^{-4}	1×10^{-4}
95th percentile	4×10^{-4}	3×10^{-4}
SOARCA UA Base Case	9×10^{-5}	3×10^{-5}

Figure 3: Combined Aleatory and Epistemic Uncertainty Conditional Individual LCF Risk (per Event) CCDF and Epistemic Uncertainty Conditional, Mean (Over Weather Variability), Individual LCF Risk (per Event) CCDF for the Specified-Radius Circular Areas



Comparing the spread of results in Figure 2 for source term (releases) versus Table 3 for LCF risk indicates that health-effect risks don't vary as much as source term (a companion paper at this conference, "SOARCA Peach Bottom Atomic Power Station Long-Term Station Blackout Uncertainty Analysis: Contributions to Overall Uncertainty," has more details about the relative contributions to overall uncertainty). This is because the long-term exposure dominates LCF risk. The long-term dominates the overall health effect risk within the 10-mile emergency planning zone (EPZ) because the emergency response is faster than the onset of environmental release. More than half the time, the long-term is the

[2] Note that the scenario frequency is ~3 x 10^{-6}/reactor-year

larger contributor to overall health effect risk beyond the EPZ. The habitability criterion controls the long-term risk because if relocated as part of protective actions, people are not allowed to return until doses are below the habitability criterion.

For the conditional, mean, individual LCF risk, within different circular areas (with 10- to 50-mile radii around the plant), the different regression techniques explain 40-85% of the variance in the results, with the recursive partitioning analysis consistently capturing the most variance.

All regression methods consistently rank the following parameters, respectively, as the most important input variables for LCF risk:

- The MACCS2 dry deposition velocity,
- The MELCOR SRV stochastic failure rate, and
- The MACCS2 risk factor for cancer fatalities for the residual organ[3].

The following additional variables also consistently show some level of importance at all circular areas in at least one of the regression methods:

- The MELCOR fuel failure criterion,
- The MELCOR drywell liner melt-through open area,
- The MACCS2 dose and dose-rate effectiveness factor for the residual organ.

These six variables alone account for 26%-75% of the variance in LCF risk results using the different regression methods. In other words, of the hundreds of variables included in this uncertainty analysis, a handful of variables drive most of the uncertainty in the consequence results. The MELCOR variables include those that are responsible for much of the variance in the source term (releases). The MACCS2 dry deposition velocity describes how fast contaminants deposit on the ground, and groundshine is the major contributor to long-term doses. While wet deposition (during precipitation events) more rapidly deposits contaminants on the ground, the wet deposition parameters are not as important because precipitation occurs only ~7% of the time at the Peach Bottom site. The MACCS2 risk factor for cancer together with the dose and dose-rate effectiveness factor determine how fatal a given dose is in the calculation of LCF risk.

Unlike the SOARCA analyses [1, 2], a nonzero early-fatality risk was calculated for some MACCS2 realizations. 13% of the 865 MACCS2 realizations calculated a nonzero early-fatality risk. The largest value of the mean, acute dose for the closest resident (i.e., 1.6 to 2.1 kilometers from the plant) for most of the source terms is about 0.3 gray (Gy) to the red bone marrow, which is usually the most sensitive organ for early fatalities. The acute dose threshold to the red bone marrow has 5th percentile and median values of about 1.1 and 2.3 Gy, respectively, in the distribution used in this study. Table 4 shows statistical results for conditional, mean, individual, early-fatality risk (per event) from the uncertainty analysis at the specified circular areas. A non-zero calculated result is possible only by combining a small percentage of the epistemic samples with a small percentage of the weather trials. After considering the scenario frequency, the calculated absolute individual early-fatality risk is on the order of 10^{-12}/reactor-year for the closest-in non-evacuating residents.

---

[3] The residual organ is represented by the pancreas and is used to define all latent cancers not specifically accounted for in the MACCS2 model. The pancreas is chosen to be a representative soft tissue.

Table 4. Conditional[4], Mean (Over Weather Variability), Individual Early-Fatality Risk Statistics for Specified-Radius Circular Areas

	0-1.3 miles	0-2 miles	0-3 miles	0-5 miles	0-10 miles
Median and 75th percentile	0.0	0.0	0.0	0.0	0.0
Mean	$5x10^{-7}$	$2x10^{-7}$	$6x10^{-8}$	$1x10^{-8}$	$5x10^{-9}$
95th percentile	$2x10^{-6}$	$7x10^{-7}$	$5x10^{-10}$	0.0	0.0

5. METHODOLOGICAL INSIGHTS

In explaining the variations in possible source terms and consequences, the use of more advanced regression techniques proved to be advantageous because they capture interaction effects and non-monotonic effects missed by the linear rank regression technique. Interaction effects among variables and non-monotonic effects are common in complex systems, such as nuclear power plant systems and environmental factors during and after a severe accident. Furthermore, the use of select single-realization analyses (analyzing the results of one Monte Carlo sample at a time) in this uncertainty analysis proved useful in validating the results of the statistical regression analyses through phenomenological explanations.

The following companion papers in this conference provide more in-depth details of results and insights from this uncertainty analysis: (1) "SOARCA Peach Bottom Atomic Power Station Long-Term Station Blackout Uncertainty Analysis: Knowledge Advancement," (2) "SOARCA Peach Bottom Atomic Power Station Long-Term Station Blackout Uncertainty Analysis: Convergence of the Uncertainty Results," and (3) "SOARCA Peach Bottom Atomic Power Station Long-Term Station Blackout Uncertainty Analysis: Contributions to Overall Uncertainty."

6. CONCLUSIONS

This uncertainty analysis [4] corroborates the SOARCA project (NUREG-1935) conclusions with the following:

- Public health consequences from severe nuclear accident scenarios modeled are smaller than those projected in NUREG/CR-2239.
- The delay in releases calculated provide more time for emergency response actions (such as evacuating or sheltering).
- "Essentially zero" absolute early fatality risk is projected:
  - The mean absolute early fatality risk is on the order of 10^{-12} per reactor year[5] within 1 mile of the plant boundary, and even this minute risk is based on less than 13% of 865 samples having a non-zero calculated risk; 87% had zero risk.
- The long-term dominates the overall health effect risk within the 10-mile emergency planning zone (EPZ) because the emergency response is faster than the onset of environmental release. More than half the time, the long-term is the larger contributor to overall health effect risk beyond the EPZ.

---

[4] These should be multiplied by the scenario frequency to calculate absolute risk. In SOARCA [1], the LTSBO scenario frequency is estimated to be ~$3x10^{-6}$ per reactor-year.

[5] Estimated risks below 10^{-7} per reactor year should be viewed with caution because of the potential impact of events not studied in the analyses, and the inherent uncertainty in very small calculated numbers.

- A major determinant of source term magnitude is whether the sticking open of the SRV occurs before or after the onset of core damage. Compounding this effect is whether or not main steam line creep rupture occurs, which leads to higher consequences.
- Health-effect risks don't vary as much as source term (releases) because people are not allowed to return until doses are below the habitability criterion.
- This analysis confirms the known importance of some phenomena (for example, the dry deposition velocity in MACCS2), and reveals some new phenomenological insights (for example, the importance of the drywell liner melt-through area in MELCOR).
- The use of multiple regression techniques provides better explanatory power of which input parameters are most important to uncertainty in results.

The results and insights from this uncertainty analysis are expected to be useful for on-going and future work, such as informing the technical bases for post-Fukushima regulatory activities and the NRC's Site Level 3 PRA project. This uncertainty study adds to the body of knowledge created by the SOARCA project, through the generation of 865 variations of how an LTSBO scenario may evolve a BWR. This study is already informing some NRC activities. For example, the spread of LCF risk results from this uncertainty analysis was used to lend confidence to the projected spread of consequence results in the uncertainty analysis supporting SECY-12-0157, "Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments" [7]. Other envisioned uses of this work are to identify key sources of uncertainty (per NUREG-1855 [8] guidance on treatment of uncertainty) for the Level 2 portion of PRA studies for BWR Mark I plants, and the Level 3 portion of PRA studies for light-water reactor severe accidents. The results also identify areas where improving our state-of-knowledge, or our state-of-modeling capabilities, could significantly reduce uncertainties in outcomes. Examples of this are improving our knowledge of BWR SRV behavior under severe accident conditions, and improving our knowledge and modeling of off-site contaminant deposition velocities. This analysis also confirms the importance of using more advanced regression techniques, such as recursive partition analysis, for identifying important inputs (and their joint influences) in complex uncertain systems.

Acknowledgements

Sandia National Laboratories is a multi-program laboratory managed and operated by Sandia Corporation, a wholly owned subsidiary of Lockheed Martin Corporation, for the U.S. Department of Energy's National Nuclear Security Administration under contract DE-AC04-94AL85000. The U.S. Nuclear Regulatory Commission sponsored and participated in this work.

References

[1] U.S. Nuclear Regulatory Commission (NRC), NUREG-1935, "*State-of-the-Art Reactor Consequence Analyses (SOARCA) Report*," NRC, 2012, Washington, DC.
[2] NRC, NUREG/CR-7110 Volume 1 – Revision 1, "*State-of-the-Art Reactor Consequence Analyses Project Volume 1: Peach Bottom Integrated Analysis*," NRC, 2013, Washington, DC.
[3] NRC, NUREG/CR-7110 Volume 2 – Revision 1, "*State-of-the-Art Reactor Consequence Analyses Project Volume 2: Surry Integrated Analysis*," NRC, 2013, Washington, DC.
[4] NRC, Draft NUREG/CR-7155, "*State-of-the-Art Reactor Consequence Analyses Project – Uncertainty Analysis of the Unmitigated Long-Term Station Blackout of the Peach Bottom Atomic Power Station - DRAFT*," NRC, 2013, Washington, DC.
[5] Bixler, N.E., et al., NUREG/CR-7161, "*Synthesis of Distributions Representing Important Non-Site-Specific Parameters in Off-Site Consequence Analysis*," NRC, 2013, Washington, DC.

[6] NRC, NUREG/CR-2239, "*Technical Guidance for Siting Criteria Development*," NRC, 1982, Washington, DC.
[7] NRC, SECY-12-0157, "*Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments,*" NRC, 2012, Washington, DC.
[8] NRC, NUREG-1855, "*Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making,*" NRC, 2009, Washington, DC.