FINDINGS FROM UNCERTAINTY STUDIES EVALUATING SEVERE ACCIDENT PHENOMENA AND OFF-SITE CONSEQUENCES USING MELCOR AND MACCS

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Abstract. The State-of-the-Art Reactor Consequence Analyses (SOARCA) project published best estimate analyses for select accident scenarios at the Peach Bottom Atomic Power Station and Surry Power Station in 2013 [1, 2, 3]. This work was followed by a first of a kind integrated uncertainty analysis (UA) performed on the SOARCA unmitigated long term station blackout (LTSBO) scenario for Peach Bottom [4]. The approach developed for the Peach Bottom UA was implemented in the Surry Power Station and Sequoyah Nuclear Plant UAs for an unmitigated short-term station blackout (STSBO) scenario [5, 6]. These UA projects included an integrated Monte Carlo analysis using the MELCOR and MACCS (MELCOR Accident Consequence Code System) codes [8, 11]. Regression analyses and separate sensitivity analyses were conducted to understand the contributions of individual uncertain inputs to the uncertainty in analysis results. Single realizations were analyzed to investigate individual parameter effects, typically associated with the extreme bounds, and to confirm phenomenological explanations of variations in system behavior and results. Consistent with the UAs, rank regression, quadratic regression, recursive partitioning, and multivariate adoptive regression splines (MARS) techniques were used to identify the importance of the input parameters with regard to the uncertainty of the results. These analyses provide insights into the sensitivity of the best estimate results to selected model inputs, corroborate the conclusions of the original SOARCA study, and further extend the body of knowledge on severe reactor accidents.

Keywords Uncertainty Analysis, Sensitivity Analysis, Severe Accidents, MELCOR, MACCS

1. INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) initiated the SOARCA project to develop current realistic evaluations of the offsite radiological health consequences for potential severe reactor accidents for two pilot plants: The Peach Bottom Atomic Power Station, a boiling-water reactor (BWR) in Pennsylvania, and the Surry Power Station, a pressurized-water reactor (PWR) in Virginia. The SOARCA project evaluated plant improvements and changes not reflected in earlier NRC publications from 1975-1990.

This paper describes the integrated UAs for the SOARCA project that were directed by the NRC and conducted by Sandia National Laboratories (SNL). These UAs evaluate the SOARCA unmitigated LTSBO scenario for Peach Bottom and unmitigated STSBO scenarios for Surry and Sequoyah (PWRs) [4, 5, 6]. The analyses used the existing SOARCA models implemented in the MELCOR code for accident progression and release analysis, the MACCS code for offsite consequence analysis, and a representative set of important uncertain parameters. The UAs used NRC, industry, and experimental data and expert judgment supplemented with limited external peer review to select a set of uncertain

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parameters to include in each study, and distributions representing the current state-of-knowledge for each parameter. The uncertainty in these parameters was then propagated to release and offsite radiological health consequence results (i.e., individual latent cancer fatality risk and individual early fatality risk) using a two-step Monte Carlo simulation process. A variety of techniques were used to examine the results including regression analyses, study of select individual Monte Carlo samples, scatter plots, and supplemental separate sensitivity analyses. The approach employed by SNL and the NRC within the SOARCA project is a case study for a comprehensive uncertainty analysis methodology to quantify the uncertainty in simulation results, and the relative impact of uncertain model parameters on predicted accident consequences.

Computer codes, such as MELCOR [7, 8, 9] and MACCS [10, 11, 12], dynamically model the initiation and progression of a reactor accident (MELCOR) through potential radiological release and atmospheric dispersion, as well as environmental and health consequences (MACCS). Severe accident modeling represents the convergence of a large number of situational variants and uncertain inputs. Uncertainty and sensitivity analysis prove instrumental in producing useful and informative simulations. MELCOR is a modern system-level analysis code which models severe reactor accident phenomena including thermal-hydraulic response; core heat-up, degradation, and relocation; hydrogen production; transportation; combustion; and fission product release and transport behavior. MACCS modeling capabilities represent important aspects of radionuclide atmospheric transport, emergency response, and dose response to radiation exposure.

2. PEACH BOTTOM UNCERTAINTY ANALYSIS INSIGHTS

The Peach Bottom unmitigated LTSBO uncertainty analysis (UA) [4] confirmed some known insights, and also uncovered new insights. For example, while it was recognized in the original SOARCA Peach Bottom best estimate analysis that a MSL rupture may occur, and a separate sensitivity analysis was performed, the range of conditions that lead to MSL rupture was not fully appreciated until the completion of the Peach Bottom UA. Performing the source term calculations of the Peach Bottom UA revealed three groupings of similar accident progression sequences within the Peach Bottom unmitigated LTSBO scenario from 865 realizations:

1. Early stochastic failure of the cycling safety relief valve (SRV), which was the deterministic SOARCA scenario in NUREG-1935 [1];
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 sequence groups exhibited differences in release magnitude, with MSL failure generally leading to the largest environmental releases. The majority of samples in this uncertainty analysis resulted in larger iodine and cesium releases than the SOARCA project calculated [2] because early stochastic failure of the cycling SRV generally leads to smaller releases. The accident progression path depended on the values sampled for a couple of key uncertain variables: the SRV stochastic failure rate (the rate at which the SRV fails to close on demand, thereby remaining fully open), and the SRV open area fraction if the SRV fails thermally. The data supporting the uncertainty distributions for these two variables is sparse due to the low frequency of operational failures and infeasibility of duplicating severe accident conditions in experiments. Similarly, there was considerable uncertainty in the selection of appropriate distributions for other important variables, such as the size of the opening that results from core melt contacting and failing the drywell liner. As a result, this uncertainty analysis was 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.

Another unexpected insight from the Peach Bottom UA was that sometimes, it was more beneficial, from the perspective of lower cesium releases, to have cesium specified as cesium hydroxide rather than cesium molybdate. This was unexpected because conventional thinking was that cesium molybdate

would result in lower releases because of lower vapor pressures. However, SME investigation into individual realizations revealed that cesium hydroxide was getting chemisorbed and trapped on RPV internals if temperatures were sufficiently high (existing MELCOR modeling of a confirmed physical phenomenon), introducing a temperature dependency on which chemical form is beneficial in a given accident progression.

Overall, several influences were found to strongly affect the magnitude of the source term (the magnitude of cesium and/or iodine releases); the following were found to be influential:

- Whether the SRV sticks open 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 MSL creep rupture occurs (largely determined by the two SRV factors above), with higher releases if MSL failure occurs due to fission products being vented straight to the drywell and thus 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 for high-temperature scenarios such as MSL creep 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 wetwell water surge leading to larger releases.

Figure 1 shows the fraction of the iodine core inventory released to the environment over time.
Figure 2 shows the fraction of cesium core inventory released to the environment over time. The earliest releases in this UA began after 10 hours, which provided sufficient time to implement planned protective actions.

Table 1 shows the distribution of results for the conditional, mean, individual latent cancer fatality (LCF) risk assuming the linear-no-threshold (LNT) dose-response hypothesis.

All of the regression methods used in the analysis consistently ranked the following parameters as the most important input variables for LCF risk:

1. The MACCS dry deposition velocity,
2. the MELCOR SRV stochastic failure rate, and
3. the MACCS risk factor for cancer fatality for the 'residual' organ[2].

The following additional variables also consistently showed 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 MACCS dose and dose-rate effectiveness factor for the 'residual' organ.

The MELCOR variables indicated as significant included those that were responsible for much of the variance in the source term (environmental releases). The MACCS dry deposition velocity describes how fast radionuclides deposit on the ground, and groundshine is the major contributor to long-term doses. While wet deposition (during precipitation events) more rapidly deposits radionuclides on the ground, the wet deposition parameters are not as important because precipitation occurs only ~7% of

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

the time at the Peach Bottom site. The MACCS risk factor for cancer together with the dose and dose-rate effectiveness factor determine the potential lethality of a given dose assuming the LNT dose response model.

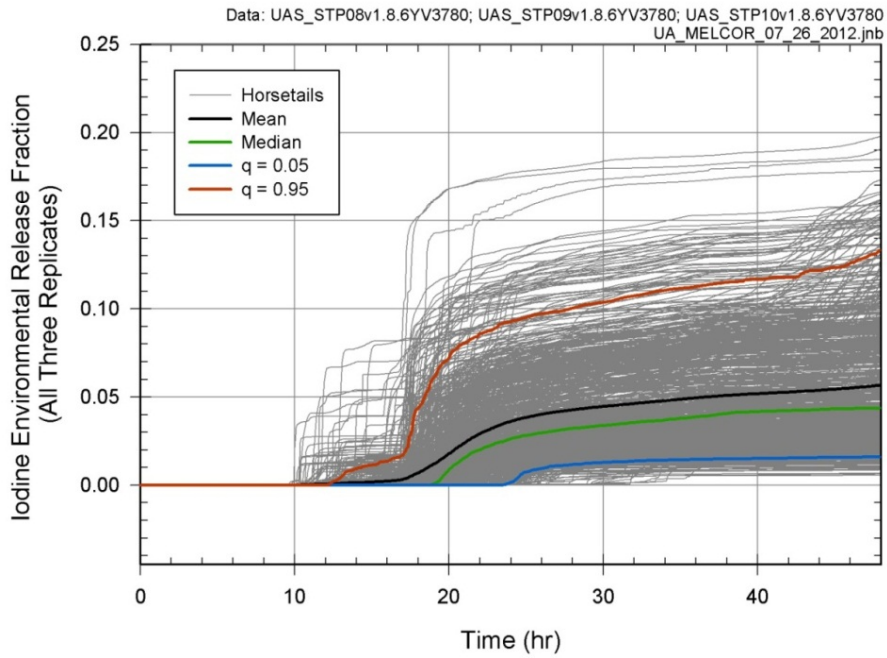

Figure 1 Peach Bottom UA iodine environmental release fraction for 48 hours.

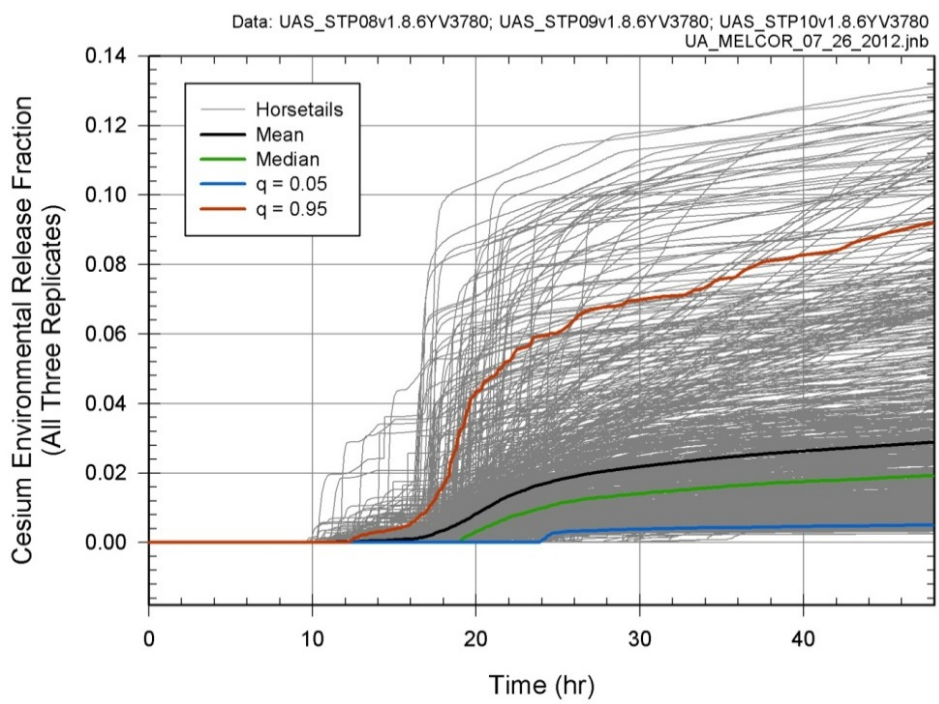

Figure 2 Peach Bottom UA cesium environmental release fraction for 48 hours.

Table 1 Peach Bottom UA conditional, mean[3], individual LCF risk (per event) averaged statistics for circular areas around the plant.

	0-10 miles	0-20 miles	0-50 miles
5th percentile	3×10^{-5}	5×10^{-5}	2×10^{-5}
Median	1×10^{-4}	2×10^{-4}	7×10^{-5}
Mean	2×10^{-4}	3×10^{-4}	1×10^{-4}
95th percentile	4×10^{-4}	8×10^{-4}	3×10^{-4}
Peach Bottom UA Reference Case	9×10^{-5}	8×10^{-5}	3×10^{-5}

2.1 Conclusions from the Peach Bottom Uncertainty Analysis

In explaining the variations in possible source terms and consequences, the use of more advanced non-linear regression techniques (quadratic regression, recursive partitioning, and multivariate adaptive regression splines) 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 proved useful in validating the results of the statistical regression analyses through phenomenological explanations.

The uncertainty analysis documented in NUREG/CR-7155 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.
- 'Essentially zero' absolute early fatality risk is projected.
- The long-term phase dominates the overall health effect risk within the 10-mile emergency planning zone (EPZ) because the emergency response is expected to be effective prior to the onset of environmental release.
- A major determinant of source term magnitude is whether the SRV sticks open before or after the onset of core damage. Compounding this effect is whether or not MSL creep rupture occurs, which leads to higher environmental releases and consequences.
- Health-effect risks do not vary as much as the source terms (environmental releases) because people are not allowed to return until doses are below the habitability criterion.
- This analysis confirms the known importance of some phenomena (e.g., the dry deposition velocity in MACCS), and reveals some new phenomenological insights (e.g., 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 the results.

3. SURRY UNCERTAINTY ANALYSIS INSIGHTS

The draft Surry unmitigated STSBO uncertainty analysis[4] [5] was based on 1003 realizations. One of the more interesting aspects of the study included modeling the conditions that could induce a steam generator tube rupture (SGTR) due to the SBO. In the draft Surry UA, an SGTR occurred in 10% of realizations, and a hot leg nozzle rupture occurred in the other 90%; in every realization that an SGTR occurred, a hot leg nozzle rupture also occurred. The majority of the realizations also had a failure to close of one or more safety valves (SVs) on the secondary and primary sides.

[3] The 'mean' within this context is in reference to the expected value over sampled weather conditions representing a year of meteorological data and over the entire residential population within a circular region. This is also applicable for early-fatality risk results.

[4] The Surry uncertainty analysis is being updated and the revised report is forthcoming in 2018.

The uncertainty analyses produced sets of time-dependent results (e.g., horsetail plots) which were used in the analysis. Figure 3, shows the horsetails for cesium release over the 48-hour analysis period. A wide spread is observed between the calculated mean, median, and 95th percentile curves. The 95th percentile falls within the population of SGTR realizations (those in the upper portion of Figure 3), the median falls within the non-SGTR realizations, while the mean is in between the two where no actual realizations exist. There is a significant time difference between the mean and median, with the median release beginning much later in time. This is an artifact of the pointwise calculations of mean and median over a bimodal population. Because the SGTR releases are about two orders of magnitude higher than the other runs, they disproportionally influence the mean, which is not representative of any specific realization.

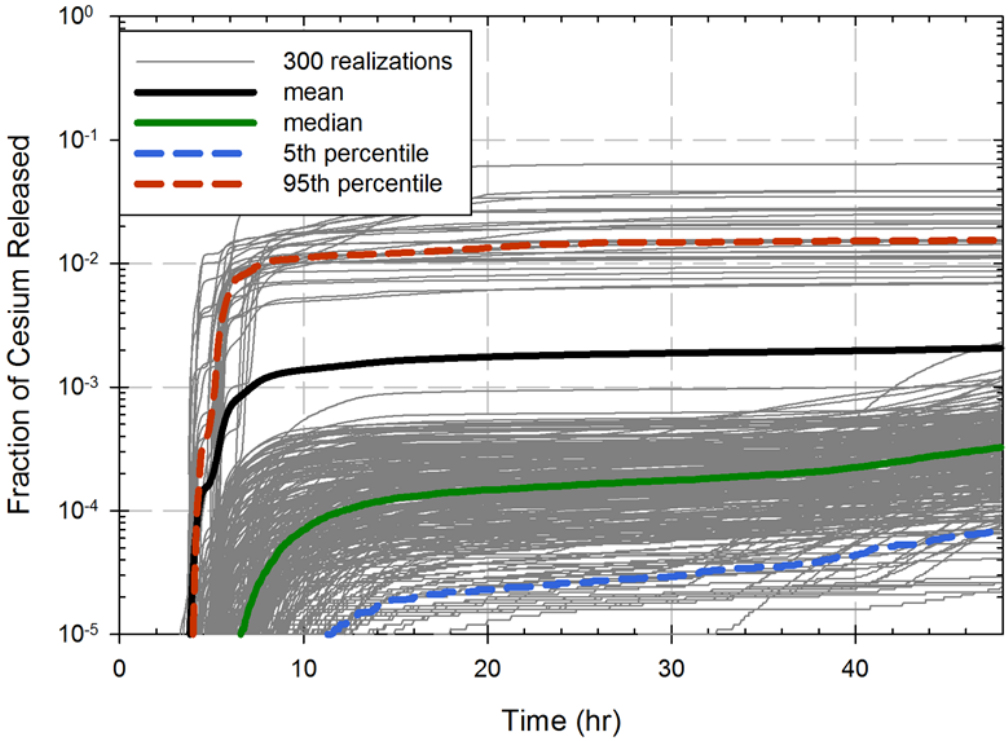

Figure 3 Surry UA cesium environmental release fraction for 48 hours.

The most influential parameters contributing to cesium and iodine environmental release fractions for non-SGTR are time at cycle (e.g., burnup), nominal leakage, the containment failure curve, and dynamic shape factor. The first three influence containment pressurization rates and determine the open area from the containment to the environment, while dynamic shape factor influences agglomeration and deposition rates for aerosols before release. The time at cycle parameter is specific to Surry and very well understood, while nominal leakage and the dynamic shape factor are based on technical specifications and experimental results, respectively. There is slightly lower confidence in the containment failure curve (CFC) model, based on difficulties scaling from the 1/6th scale tests from which the model was created, but the yield-before-rupture behavior was confirmed as more realistic by structural experts. Important for iodine release fractions but not cesium was the fraction of gaseous iodine, because it does not deposit on structures but remains airborne. Thus, it is released in much higher percentages than aerosolized cesium iodine (CsI). The two parameters that determine the amount of gaseous iodine, time at cycle and chemical form of iodine, are both identified as significant main contributors by the regression analyses.

The two parameters that show the highest importance in determining whether an SGTR occurs are the initial tube thickness of the most damaged steam generator tube in the hottest region, and the fraction of

the full open area of a primary or secondary SV at the time it fails. The tube thickness parameter directly affects the initial damage state of one of the steam generator tubes. The SV open area parameter influences the depressurization of either the primary or secondary after SV failures to close, and thus controls the pressure differential across the damaged steam generator tube. Testing of SGTR realizations, through one-off calculations showed that a pressure differential of 1000 psi or more was needed during core damage to induce the SGTR. Additionally, the magnitude of releases for an SGTR is primarily driven by the time between the SGTR and hot leg creep rupture, because before hot leg rupture, the SGTR is the primary transport path for radionuclides released from the fuel.

The MACCS results were generated assuming the LNT dose-response hypothesis. Like the original SOARCA study [1, 3], the Surry UA demonstrates that early fatality risks are negligible, essentially zero, and LCF risks are even lower than those evaluated in the original SOARCA study. Table 2 shows the mean LCF risks from this UA.

Table 2 Surry UA conditional, mean, individual LCF risk (per event) averaged statistics for specified radial distances around the plant.

	0-10 miles	10-20 miles	0-50 miles
UA Mean	3×10^{-5}	1×10^{-5}	5×10^{-6}
UA Median	6×10^{-6}	2×10^{-6}	9×10^{-7}
UA 5th percentile	7×10^{-7}	2×10^{-7}	1×10^{-7}
UA 95th percentile	2×10^{-4}	7×10^{-5}	4×10^{-5}
SOARCA estimate, STSBO	1×10^{-4}	N/A	2×10^{-5}
SOARCA estimate, STSBO with SGTR	3×10^{-4}	N/A	7×10^{-5}

The most influential input parameters were identified with respect to the figures of merit:

- For releases in non-SGTR realizations, the most important parameters were time within the fuel burnup cycle, containment design leakage, the containment failure curve, and the dynamic shape factor for radionuclide particles. For iodine, the amount that was assumed to be gaseous was important, which is determined by time within the fuel burnup cycle and chemical form of iodine.
- Primary SV open fraction and tube thickness were the main determinants regarding whether an SGTR occurred, and secondary SV open fraction had the highest importance in cesium and iodine environmental release fractions for SGTR releases.
- For LCF risk, steam generator tube thickness and SV open area fraction are the most influential parameters. These two parameters largely determine whether the accident progresses toward a SGTR. Thus, they have an important influence on magnitude and timing of the environmental release which directly influences LCF risk.
    - Three other parameters have an effect on LCF risk. These are time within the fuel burnup cycle, groundshine shielding factor, and containment design leakage.
- The most important parameter in the Peach Bottom UA [4], dry deposition velocity, is not shown to be very important in the Surry UA. This may be because the distribution for dry deposition velocity was refined for the Surry UA or it may be because some other parameters are relatively more important (e.g., the parameters that influence the occurrence of SGTR). The rationale for the narrower distribution is that the one used in the Peach Bottom UA reflects variations from one weather instance to another, not variations in the best value to use for an entire year of weather data.

3.1 Conclusions from the Surry Uncertainty Analysis

The Surry UA modeled distributions for parameter values that were historically modeled with fixed values and applied multiple non-linear regression techniques to support an understanding of the results. Such an analysis produces substantial information, and a summary of important insights is provided:

- SGTRs occurred in about 10% of the realizations and had environmental release fractions one to two orders of magnitude larger than non-SGTR realizations.
- SGTRs always included both a thermal and pressure element.
- In most of the realizations, iodine and cesium environmental release fractions were higher early in the transient than the original Surry SOARCA calculation [3], but all were lower at 48 hours, except that cesium was equal in a few realizations.
- Lower release fractions at 48 hours were primarily driven by time at cycle sampling, higher nominal containment leakage, and changes to the containment failure model (i.e., gradually degrading containment versus sudden catastrophic failure). All of these updates lead to slower containment pressurization and the leak-before-break failure modeling prevents large amounts of revaporization.
- The LCF risk was observed lower than the original Surry SOARCA calculation [3], and is attributable to the lower source terms from the UA (again due to more realistic containment degradation modeling).
- The consequence analysis showed that the LCF risk distribution is much narrower when only uncertain consequence parameters are considered than when both source-term and consequence parameters are considered in the analysis. It appears the results are more heavily influenced by uncertainties in source term than by uncertain consequence parameters, just as they were for the Peach Bottom [4]. While this is true when only LNT is considered, uncertainties in LCF risks are created by uncertainties in the type of dose-response model used (e.g., dose truncation models), and most likely would alter this conclusion, if dose response had been included as part of the integrated uncertainty analysis.
- In the *number of steam generator tubes* joint sensitivity analysis, one realization with five tubes failing had no hot leg creep rupture failure leading to the highest release fractions.

4. SEQUOYAH UNCERTAINTY ANALYSIS INSIGHTS

The Sequoyah SOARCA analyses [6] focused on insights unique to the ice condenser containment and potential vulnerabilities to hydrogen challenges, and insights are based 567 realizations. Ice condenser containment-specific and hydrogen-specific considerations were added as uncertain parameters.

This analysis indicates that if containment does not rupture early following the first burn (within 12 hours), subsequent hydrogen burns are not energetic enough to rupture the steel containment vessel later in the sequence because of decreasing hydrogen concentrations. Due to consumption of oxygen during burns, ongoing vaporization of ice melt, and non-condensable gas generation from core-concrete interactions (CCI), the reduced oxygen concentration in containment is observed to the point where it is insufficient to support further burning. Deflagrations cease but containment continues to pressurize as fission product decay heat incessantly drives vaporization of ice melt and non-condensable gas generation. The pressurization is monotonic and most often pressurizes containment to rupture prior to 72 hours (end of simulation time). Notable exceptions to this are realizations having beginning of cycle (BOC) representations of the Sequoyah reactor core. None of the BOC realizations over pressurized containment by 72 hours due to the lower decay heat. Additionally, the sampled value of containment rupture pressure was seen as crucial to whether or not containment fails early.

Of paramount importance to whether or not an early containment failure occurs is the amount of hydrogen that accumulates in the containment dome prior to the first deflagration. In order for this to happen, much of the hydrogen generated in-vessel must migrate to the containment dome and be burned there. In interpreting this result, the following should be considered:

1. For this study, burns are allowed to originate only in the lower containment, and therefore for burning to occur in the dome, a burn must first have ignited in lower containment and subsequently propagate to the dome.
2. Burns are ignited only by:
    a. Hot gases issuing from a hot leg breach,

   b. Hot gases issuing from the PRT, or
   c. Core debris on the containment floor.
3. Pressures developed in the upper containment, which are from burns ignited in the lower containment and subsequently propagate to the dome, can be markedly higher than pressures developed in lower containment.
4. The containment vessel has been determined to be weakest at the equipment hatch and thus containment rupture is triggered on excessive pressure in upper containment as opposed to excessive pressure in lower containment.

Early ignition of a hydrogen deflagration by hot gasses issuing from the PRT can preempt the accumulation of hydrogen in the containment dome such that a large deflagration, which otherwise would have failed containment, does not materialize. However, this is not always the case.

The rupture pressure, time within the fuel burnup cycle, and melting temperature of the eutectic formed between ZrO_2 and UO_2 are the most influential parameters on the time of rupture. The aggregate primary SV cycles for a three-valve system failure to close is indicated as having a strong conjoint influence and effectively no main influence for the non-linear regressions. The rupture time is highly dependent upon time within the fuel burnup cycle, with the beginning of cycle (BOC) realizations failing much later than either middle of cycle (MOC) and end of cycle (EOC) realizations. The difference between MOC and EOC realizations is comparatively less significant.

The time within the fuel burnup cycle is significant for both cesium and iodine environmental release, varying from 'nearly zero' for BOC realizations and then increasing in magnitude and decreasing in timing between MOC and EOC conditions. The aggregate number of primary SV cycles for a three-valve system failure to close is also significant, and the results are consistent with deterministic analyses [6]. The primary SV cycle parameter also has high interaction effects for both the cesium and iodine non-linear regressions. No other parameters were identified as having significant effect on cesium or iodine environmental release. Generally, cesium and iodine environmental release is minimal until about 42 hours into the simulation, and increases significantly from 48 hours to 72 hours, as seen in Figure 4; RLZ 266 is the reference realization, selected based upon near-median values for all figures of merit, and the 5th percentile falls below 10^{-5} environmental release fraction.

The MACCS model was developed assuming the large seismic initiating event affects the evacuation routes. This Sequoyah SOARCA approach differs from earlier Peach Bottom [2, 4] and Surry [3, 5] SOARCA efforts with regard to the state of infrastructure assumed. In those analyses, impacts on evacuation road networks and infrastructure were considered in sensitivity analyses rather than as part of their respective UAs. Sequoyah roadway access and capacity are affected by the assumption that bridges in the 10-mile EPZ are unusable. The infrastructure beyond the Sequoyah EPZ is assumed to be unaffected by the earthquake. Since it is difficult to consider all potential scenarios with respect to damage incurred within the EPZ, the primary factors modeled in this study are the evacuation speeds and delays due to the loss of roadways with bridges.

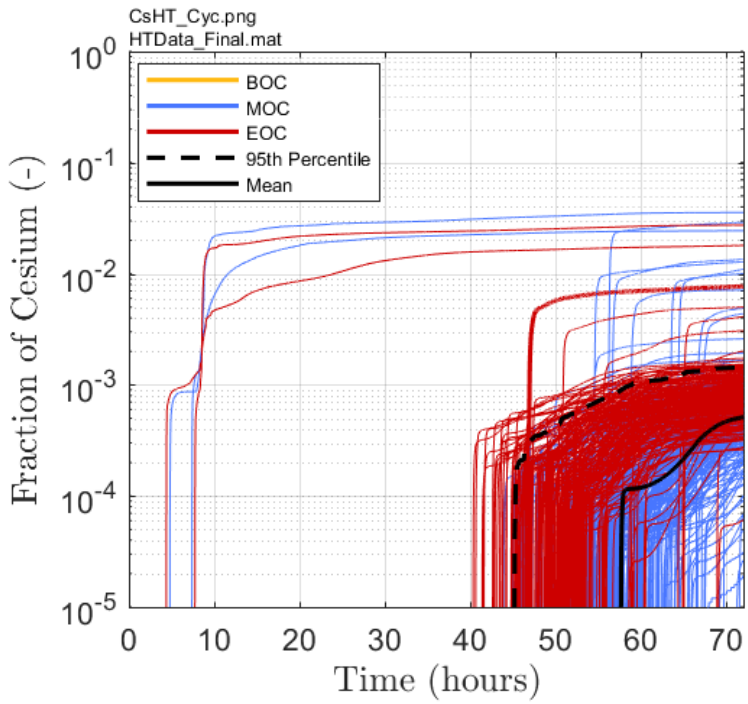

Figure 4 Sequoyah UA cesium environmental release fraction for 72 hours.

Assuming the LNT dose-response, LCF risks generally decrease with increasing distance from Sequoyah; see Table 3. Contributions from the long-term phase risks dominate the emergency-phase risks for the large majority of the realizations. Similar to the results from the Peach Bottom and Surry SOARCA UAs [4, 5], the Sequoyah analyses show essentially zero early fatality risk, and a low LCF risk for the affected population. Even for STSBO variations leading to early containment failure, release to the environment begins prior to the completion of the 10-mile EPZ evacuation. LCF risk calculations are generally dominated by long-term exposure to small annual doses (below 2 rem in the first year after the accident and below 500 mrem per year in subsequent years corresponding to the habitability criterion) for populations exposed to residual contamination over a long period of time.

Table 3 Sequoyah UA conditional, mean, individual LCF risk (per event) averaged statistics for specified radial distances around the plant.

	0-10 miles	10-20 miles	0-50 miles
Mean	8×10^{-5}	1×10^{-4}	9×10^{-5}
Median	7×10^{-5}	8×10^{-5}	8×10^{-5}
5th percentile	1×10^{-8}	3×10^{-9}	2×10^{-9}
95th percentile	2×10^{-4}	3×10^{-4}	2×10^{-4}

Regression analyses indicate that the time within the fuel burnup cycle when the accident occurs has the largest influence on consequences. Within the EPZ, three MACCS parameters and two additional MELCOR parameter are also assessed as important. The MACCS parameters are the cancer risk factors for residual and colon organs, and the long-term groundshine shielding factor. The cancer fatality risk factor for the 'residual' organ[5] represents all of the cancer types not specifically treated in MACCS. The MELCOR parameters are the aggregate primary SV cycles and containment rupture pressure. Two additional MACCS parameters are important at further distances presented in the regression analysis; the cancer risk factor for lung cancer and normal relocation time.

[5] MACCS uses eight cancer sites (organs), seven of which are specific (lung, red bone marrow, bone, breast, thyroid, liver, and colon) and the last of which (residual) represents the cancers not explicitly modeled and is based on the dose for the pancreas, which is used as a surrogate for other soft tissues.

Specific to the MELCOR parameters, aggregate primary SV cycles is significant with respect to uncertainty in individual LCF risk for the 0-10 and 10-20 mile ranges, but not for ranges beyond those. This parameter influences hydrogen buildup in containment and the potential for early containment failure, leading to early releases. An early release has the potential to affect evacuees within the EPZ and shadow evacuees from 10 to 15 miles because some of the evacuees can be directly affected by the plume. This parameter has a lesser influence on the non-evacuating population who are assumed to remain in place at the start of release regardless of release timing.

The time within the fuel burnup cycle is consistently significant at all distances and is driven by its strong influence on the magnitude of the source term. Consequences are more severe as this parameter increases, but the differences are more profound between BOC and MOC/EOC and less profound between MOC and EOC.

Containment rupture pressure is also significant nearer the plant, but not significant within the 30 to 40 and 40 to 50 mile ranges. The pressure at which containment ruptures is correlated negatively with consequences, which means consequences decrease as containment failure pressure increases. Lower containment failure pressure generally corresponds to earlier containment failure. Correspondingly, a higher failure pressure translates to a delay in containment failure timing which benefits both evacuation timing as well as airborne fallout effectiveness within the containment.

Specific to the MACCS parameters, the long-term groundshine shielding factor is only indicated as significant in the 10 mile EPZ. This parameter is a factor in the equation for groundshine dose, so risk increases with this factor. The cancer risk factors have high significance with the colon and residual factors consistently indicated as significant, though in differing orders at specified radial distances; these are the two largest of the LCF risk factors and both appear as multipliers in terms of the equation for individual LCF risk.

4.1 Conclusions from the Sequoyah Uncertainty Analysis

For all MELCOR output figures of merit (i.e., containment rupture timing, environmental cesium and iodine release, and in-vessel hydrogen production), the greatest influence on the UA was SV behavior. While the SV failure-to-close and open-area-fraction upon failure-to-close was investigated further and distributions updated in the Sequoyah UA, large uncertainties remain with respect to the true SV failure rates and behavior during severe accidents, due to sparse relevant data.

The Sequoyah SOARCA analyses reinforce the results of past analyses [14] of ice condenser containments showing that successful use of igniters is effective in averting early containment failure. Even for scenarios resulting in early containment failure (radioactive release to the environment prior to completion of evacuation for the 10-mile EPZ), resulting LCF risks are small and early fatality risk is essentially zero.

5. CONCLUSION

These UAs corroborate the conclusions from the SOARCA study [1, 2, 3] that:

1. The public health consequences from severe nuclear accidents modeled are smaller than previously calculated;
2. delayed releases calculated provide more time for emergency response actions such as evacuation and, hence, the long-term dominates health effect risks; and
3. negligible early fatality risk is projected.

The SOARCA analyses of these three pilot plants have been useful in many ways beyond their original objectives. The SOARCA UA project results, insights, computer code models, and modeling best practices have supported NRC rulemaking, licensing, and oversight efforts as well as facilitated international cooperation and knowledge management. Additionally, the process of conducting such

detailed analyses has developed NRC staff expertise in a variety of important technical areas including severe accident progression, environmental source terms, atmospheric transport and dispersion, offsite consequence analysis, emergency preparedness and response, dosimetry, health effects, uncertainty analysis, and risk communication. The SOARCA project also identified improvements in NRC analytical tools and associated severe accident analysis methodologies, including parametric uncertainty analysis. The improvement of tools, methodologies, and of staff technical expertise improves NRC's capabilities to carry out its mission to protect public health and safety, and the environment.

Uncertainty analyses for complex systems have successfully played a central role in many applications supporting nuclear reactor safety analysis. Coupling regression methodology with a probabilistic analysis using phenomenologically-driven deterministic models allows for qualitative and quantitative measures of the model and parameter uncertainties when using MELCOR and MACCS to evaluate the system response to postulated severe reactor accident scenarios, potential release of radionuclides, and potential human health impacts. The use of multiple statistical regression models quantitatively correlates parameter uncertainty with result dependencies. It is necessary to support the understanding of uncertainties with phenomenology which ultimately builds confidence in the regression results that identify important effects due to model or parameter uncertainties.

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6. REFERENCES

1. U.S. Nuclear Regulatory Commission, "State-of-the-Art Rector Consequence Analyses (SOARCA) Report," NUREG-1935, U.S. Nuclear Regulatory Commission, November 2012.

2. Sandia National Laboratories, "State-of-the-Art Reactor Consequence Analyses Project Volume 1: Peach Bottom Integrated Analysis," NUREG/CR-7110 Volume 1, May 2013.

3. Sandia National Laboratories, State-of-the-Art Reactor Consequence Analyses Project Volume 2: Surry Integrated Analysis," NUREG/CR-7110 Volume 2, August 2013.

4. Sandia National Laboratories, "State-of-the-Art Reactor Consequence Analyses Project – Uncertainty Analysis of the Unmitigated Long-Term Station Blackout of the Peach Bottom Atomic Power Station," NUREG/CR-7155, U.S. Nuclear Regulatory Commission, September 2015.

5. Sandia National Laboratories, "State-of-the-Art Reactor Consequence Analyses Project – Uncertainty Analysis of the Unmitigated Short-Term Station Blackout of the Surry Power Station – *DRAFT*," ML15224A001, U.S. Nuclear Regulatory Commission, January 2016.

6. Sandia National Laboratories, "State-of-the-Art Reactor Consequence Analyses Report – Sequoyah Integrated Deterministic and Uncertainty Analyses - *DRAFT*," ML17156A270, U.S. Nuclear Regulatory Commission, May 2017.

7. Sandia National Laboratories, "MELCOR Best Practices as Applied in the SOARCA Project," NUREG/CR-7008, U.S. Nuclear Regulatory Commission, Washington, DC, August 2014.

8. Humphries L.L. et al., "MELCOR Computer Code Manuals, Vol. 1: Primer and Users' Guide, Version 2.1.6840," SAND2015-6691R, Sandia National Laboratories, August 2015.

9. Humphries L.L. et al., "MELCOR Computer Code Manuals, Vol. 2: Reference Manual, Version 2.1.6840," SAND2015-6692R, Sandia National Laboratories, August 2015.

10. Sandia National Laboratories, “MACCS2 Best Practices as Applied in the State-of-the-Art Reactor Consequence Analyses Project,” NUREG/CR-7009, U.S. Nuclear Regulatory Commission, August 2014.

11. Sandia National Laboratories, "Code Manual for MACCS2: Volume 1, User's Guide," NUREG/CR-6613, U.S. Nuclear Regulatory Commission, May 1998.

12. Sandia National Laboratories, “MELCOR Accident Consequence Code System (MACCS) Model Description,” NUREG/CR-4691, U.S. Nuclear Regulatory Commission, February 1990.

13. Sandia National Laboratories, “Technical Guidance for Siting Criteria Development,” NUREG/CR-2239, U.S. Nuclear Regulatory Commission, December 1982.

14. U.S. Nuclear Regulatory Commission, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants,” NUREG-1150, U.S. Nuclear Regulatory Commission, December 1990.