State-of-the-Art Reactor Consequence Analyses Project: Uncertainty Analysis of a Potential Unmitigated Short-Term Station Blackout of the Surry Nuclear Power Station

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1 BACKGROUND

The evaluation of accident phenomena and the offsite consequences of severe reactor accidents has been the subject of considerable research by the NRC over the last several decades. As a result of this research, capability exists to conduct more detailed, integrated, and realistic analyses of severe accidents at nuclear power reactors. A desire to leverage this capability to address conservative aspects of previous reactor accident analyses was a major motivating factor in the State of the Art Reactor Consequence Analyses (SOARCA) project. Through the application of modern analysis tools and techniques, the SOARCA project (NRC, 2012) developed a body of knowledge regarding the realistic outcomes of potential severe reactor accidents with best estimate analyses of selected accident scenarios at the Peach Bottom Atomic Power Station (Peach Bottom) and the Surry Power Station (Surry). The SOARCA project continued with an integrated uncertainty analysis (UA) of the unmitigated long term station blackout (LTSBO) at Peach Bottom (NRC, 2016) and the Surry integrated UA presented herein.

The SOARCA project (NRC, 2012) analyzed selected scenarios, first assuming the events proceeded without the 10 CFR 50.54(hh) mitigation measures (unmitigated), and then assuming that the 10 CFR 50.54(hh) mitigation measures were successful (mitigated). While these analyses have generally met the SOARCA project objectives, certain additional severe accident consequence analyses were warranted to expand upon the body of knowledge developed in SOARCA and to support and inform other NRC activities including the Site Level 3 Probabilistic Risk Assessment (PRA) project and Fukushima lessons learned. These additional analyses are discussed in SECY-12-0092, "State-of-the-Art Reactor Consequence Analyses – Recommendation for Limited Additional Analysis," (NRC, 2012a), where NRC staff recommended performing a UA for a severe accident scenario at Surry.

In 2013, an integrated UA was completed on the unmitigated LTSBO for Peach Bottom (NRC, 2016). The Peach Bottom study provided a quantitative analysis of the robustness of the deterministic calculation, and in the process, demonstrated the feasibility of producing integrated uncertainty results. The ability to implement an integrated UA permitted consideration of epistemic uncertainty in parameters, and to some degree aleatory uncertainty, across accident progression, release, and consequence modeling domains. The uncertainty aspect included sampling severe accident model parameters over defined distributions and performing regression analyses to identify the importance of the input parameters with regard to the uncertainty of the results. The Peach Bottom UA results are informative, but were conducted for the unmitigated LTSBO for a boiling water reactor (BWR). As with the Surry UA presented herein, the application of the results must be tempered with an understanding of the reactor type, accident scenario for which results were produced, and site specific characteristics.

2 APPROACH AND OBJECTIVES

The Surry UA followed the approach developed for the Peach Bottom UA. Lessons learned from the Peach Bottom UA and feedback from the NRC's Advisory Committee on Reactor Safeguards (ACRS) on the Peach Bottom UA were considered, as well as additional knowledge gained since the Surry best estimate calculation (NRC, 2013). One of the original objectives of the Surry UA was to quantify the robustness of the Surry best estimate unmitigated short term station blackout (STSBO) analysis. However, since the completion of the Surry SOARCA study (NRC, 2013), there have been many enhancements and updates to the state of the art in modeling severe accidents applied in that study. Changes to the severe accident and consequence codes reduce the benefit of the initial objective of directly quantifying the robustness of the original analysis. Nonetheless, this UA provided a comparison of the results of the Surry best estimate analysis (NRC, 2013) with the current, more advanced, severe accident modeling systems applied in an uncertain framework. Additional objectives included: determining whether the Surry UA results corroborate the general conclusions and insights from the original SOARCA best estimate study; developing insights into the overall sensitivity of results to uncertainty in selected modeling inputs; identifying the most influential input parameters contributing to accident progression and offsite consequences through application of an uncertainty analysis methodology; informing the NRC's Site Level 3 PRA and post-Fukushima activities including Tier 3 items.

Figures of merit were selected to support the analysis and investigation of results. The source term (MELCOR) figures of merit were the environmental release fractions of cesium and iodine, in-vessel hydrogen production, and release timing. The consequence (MACCS) figures of merit were latent-cancer fatality (LCF) risk and early fatality risk at specified distances.

The Surry SOARCA unmitigated STSBO was selected as the accident scenario in part because of the importance of station blackout scenarios and in part because accident progression occurs relatively quickly under the postulated conditions. The relatively quick accident progression provides a basis to assess the effect of offsite response parameters while the release is potentially underway. Of the scenarios selected for Surry in the SOARCA best estimate study, the unmitigated STSBO with induced SGTR was also one of the two scenarios with the highest conditional individual LCF risk (NRC, 2012).

To meet the objective of developing insights into the overall sensitivity of SOARCA results to uncertainty in selected modeling inputs, a reasonable number of modeling inputs important to the figures of merit being assessed were chosen (the project scope did not include model uncertainty). Many parameters are basic input, such as core inventory, material properties, sizes and lengths of piping, weather files, etc. Selecting parameters was an iterative process to identify those expected to influence the results.

As developed, most of the parameters characterized epistemic uncertainty and a few characterized aleatory uncertainty. Often the mode (most likely value) or median (50^{th} percentile) of the distribution corresponded to the best estimate value used in the original analysis. In an effort to represent a state of the art study, when additional or new knowledge was available, the information was considered, and this resulted in the mode of some parameters being different than the SOARCA best estimate value (NRC, 2013).

MELCOR, MelMACCS, and the MELCOR Accident Consequence Code System (MACCS) are the three primary codes used in the integrated analysis. Uncertainty in the model inputs was propagated in a two-step Monte Carlo simulation (MCS). First, a set of source terms were generated using MELCOR, sampling uncertain MELCOR inputs. Each source term was then coupled with one set of sampled values for uncertain MACCS inputs in a second MCS to generate a set of consequence metrics. Simple random sampling was used in both MCS steps, to enable discarding incomplete MELCOR realizations and the use of bootstrapping methods to estimate confidence in results. The MACCS MCS further included an inner loop of ~1,000 weather trials to represent variability due to weather, for each epistemic vector of sampled MELCOR and MACCS inputs.

The team chose MELCOR uncertain input parameters across the following domains of modeling: accident sequence, in-vessel accident progression, ex-vessel accident progression, containment behavior, chemical forms of iodine and cesium, aerosol transport and deposition. The team chose MACCS uncertain input parameters across the following domains of modeling: deposition, dispersion, shielding factors, early health effects, latent health effects, and emergency response.

The MELCOR, MelMACCS, and MACCS codes are continually enhanced, updated, and maintained as part of the NRC research program making it difficult to perform a direct comparison of the Surry UA results with the Surry NUREG/CR 7110 Volume 2 results. A comparison of the MELCOR 1.8.6 model of Surry and the MELCOR 2.1 model was conducted and is presented in an appendix to NRC's draft SOARCA Surry UA report. With the improvements in the Surry model and in the current state of knowledge since the original SOARCA modeling, it was determined that additional selected model enhancements should be added to advance the analysis to improve areas where enhancements would benefit the analysis. A baseline MELCOR calculation (base case) was then performed of the Surry unmitigated STSBO scenario. A few observations from the base case calculation include:

- An over-cycling failure to close (FTC) of the lowest set-point safety valve (SV) on the pressurizer occurred.
- A hot leg nozzle rupture occurred.
- Energetic hydrogen deflagrations occurred in containment.
- Containment design pressure and the pressure associated with liner yield were both exceeded.
- Releases to the environment of iodine and cesium at 48 hours were small, 0.07 percent and 0.03 percent of the inventory at scram, respectively, both well below the original SOARCA source term (NRC, 2012).

3 RESULTS

After completion of the updated base case, a high performance computing cluster was used to execute a Monte Carlo simulation with 1200 MELCOR runs, of which 1003 successfully completed (MELCOR runs typically fail due to convergence or timestep errors, most commonly in the COR package) the 48 hour analysis time, the run time for the batch calculations. A set of realizations was further analyzed to 72 hours and is documented as a sensitivity study in this report.

Each MELCOR run was identified with a unique realization number. In the 1003 successful calculations (i.e., realizations):

- A steam generator tube rupture (SGTR) occurred in 104 realizations (10% of realizations), and a hot leg nozzle rupture occurred in 930 realizations;
- In every realization that an SGTR occurred, a hot leg nozzle rupture also occurred;
- A failure of one or more reactor coolant system (RCS) secondary side safety valves to close occurred in 954 realizations (95% of realizations), and an SV on the RCS primary side (on the pressurizer) failed to close in 686 realizations (68% of realizations);
- The steel containment liner yielded and tore in 742 realizations (74% of realizations); and
- Containment rebar yielded (and the concrete fractured) in 72 realizations (7% of realizations).

The uncertainty analyses produced sets of time-dependent results (e.g., horsetail plots). Figure 1, shows the horsetails for Cs release over the 48 hour analysis period. A wide spread is observed between the calculated 5th percentile, median, mean, and 95th percentile curves. The 95th percentile falls in the SGTR realizations (those in the upper portion of Figure 1), the median in the non-SGTR realizations, while the mean is in between where no actual realizations exist. There is a significant time difference between the mean and median, with the median releasing much later in time. This is due to how the mean and median are calculated. 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. Meanwhile, the median takes the middle realization at each time, and is representative of a non-SGTR at all times, although not necessarily the same realization at all times. The calculated 5th percentile curve has an associated initial release time that is much later than the calculated median curve, and the 5th percentile remains under 0.01 percent at 48 hours. The figure shows the median and 5th percentile curves start increasing at a slightly higher slope around 40 hours, primarily due to containment liner failure, and in a few cases, rebar failure, both of which are driven more by pressure increase of steam in containment due to decay heat, rather than creation of additional non-condensable gases. The 95th percentile has no observable late slope increase, because it represents an SGTR (containment bypass). The mean is primarily influenced by the SGTR realizations and also does not exhibit a late increase.

Additional insights related to accident progression were obtained through investigation of selected single realizations to identify phenomena affecting the Cs and I releases to the environment and in-vessel hydrogen production. Key phenomena identified in the MELCOR single realization analysis include:

- Containment building rebar yielding and resulting concrete fracture,
- Importance in the timing of the pressure relief tank (PRT) dryout – if the PRT fails early, there was nothing of note, but if PRT failed late, (near containment failure) it was very important,
- The number of successful cycles experienced by the system of 3 parallel SVs serving the primary-side of the RCS, which was an indication of how long the RCS stayed at high pressure, and
- Chemisorption of Cs from CsOH into the stainless steel of reactor pressure vessel (RPV) internals.

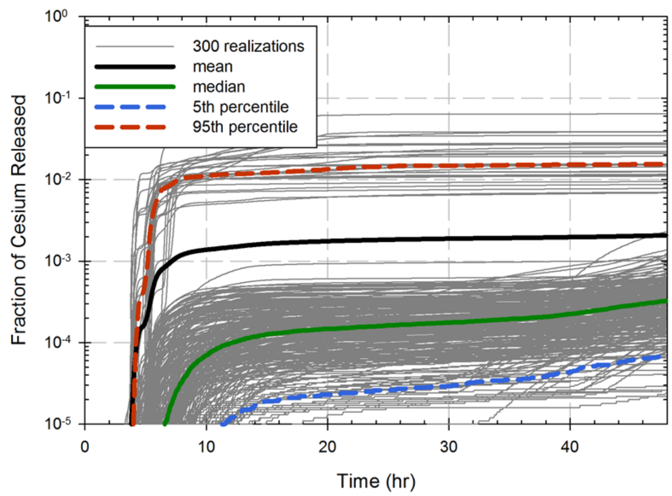

Figure 1. Cesium release fractions over 48 hours with mean, median, 5th and 95th percentiles

There were 104 SGTRs observed in the results, corresponding to a tube rupture in about 10 percent of the realizations. Regression analyses are used to determine which inputs, amongst those that are uncertain, are driving the output uncertainty. Even though only 10 percent of realizations had SGTRs, with the exception of in-vessel hydrogen production, these SGTR realizations dominated the regression results. Therefore, separate regression analyses, were performed on the full set of realizations, the set of realizations that experienced an SGTR, and the set that did not experience an SGTR.

For the full set of realizations and the non-SGTR set, regression analyses were performed for beginning of cycle (BOC), middle of cycle (MOC), and end of cycle (EOC) to understand the extent to which time at cycle (e.g., burnup) influenced non-SGTR realizations and whether any parameters would have raised importance with time at cycle kept constant. This was accomplished by including an uncertain parameter named CYCLE, which identifies the point during the fuel cycle (BOC, MOC, and EOC) at which the accident occurs. This sampled parameter directly affects the MELCOR source term calculation through decay heat, and it directly affects the MACCS consequence analysis through fission product inventory. Of all the sampled parameters, CYCLE is the only one that has such a dual status. Because fission product inventories in the fuel increase with burnup, this parameter can have a significant influence on risk. The inventories of shorter lived isotopes increase with burnup only until secular equilibrium is established; however, the inventories of longer lived isotopes, like Cs-137, can nearly double from BOC to EOC. Because the longer lived isotopes have a significant effect on LCF risk, especially in the long-term phase, this parameter is significant to the predicted results. The correlation between CYCLE and predicted risk is positive, (i.e., greater burnup relates to increased risk).

The regression analysis were performed for results obtained at the end of the 48 hour analysis period, and it should be recognized that results could be different if results were evaluated at an earlier or later time in the calculation. Results from each regression technique for the individual contribution of a parameter and the conjoint influence of a parameter on the results are provided. Conjoint influence is the influence of two or more input parameters acting together, which may have synergistic effects that would not be uncovered by studying the influence of each parameter separately and individually. Overall influence of a parameter is reported as an average of the influence suggested by the four regression techniques.

Almost all of the MELCOR realizations showed the iodine and cesium environmental release fractions were significantly lower than their respective Surry SOARCA calculation (NRC, 2013), except for a few non-SGTR realizations where cesium was equal or greater. The lower release fractions were driven primarily by slower containment pressurization, caused by a number of factors including sampling of time at cycle which impacts the total decay heat, and a modification to the containment failure model that incorporates a more realistic yield-before-rupture model.

As seen in Figure 1, there is a large split between SGTR and non-SGTR runs, where SGTR runs had one to two orders of magnitude higher release fractions. This is different than the original SOARCA analysis, primarily caused by a new model for secondary decontamination factors and a variation of the time between the SGTR and hot leg creep. The most influential parameters contributing to cesium and iodine environmental release fractions for non-SGTR (the majority of releases) 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 well understood 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 CsI. The two parameters that determine the amount of gaseous iodine, time at cycle and chemical form of iodine (ChemformI2), are both identified as significant main contributors by the regression analyses. For BOC analyses, design leakage becomes the most important parameter contributing to iodine and cesium release because containment liner yield is never reached and leakage is the only release path to the environment.

The calculated mean and median of the in-vessel hydrogen production distribution at 48 hours were similar in magnitude to the Surry SOARCA calculation 0. Regression results for hydrogen generation uncertainty were essentially the same with and without an SGTR, because this is an in-vessel effect. The most important parameters early in hydrogen production were those related to primary system depressurization, impacting steam generation and flow, followed by time at cycle. The magnitude of the total hydrogen generation within 48 hours was driven by effective fuel melt temperature and the time at cycle, both of which determine the amount of time the fuel remains in place before relocation, which suppresses the oxidation process. There were no significant differences in regression results for the time at cycle independent analyses.

For release timing, the metric evaluated was release of 1 percent of noble gas to the environment. It was observed that 18 percent of non-SGTR realizations do not release 1 percent of noble gases by 48 hours. Investigation of the release timing identified a large timing difference between SGTR and non-SGTR realizations, with non-SGTR realizations meeting the environmental release timing metric about 20 to 40 hours later, if they meet the criteria at all. In the original SOARCA analysis the timing difference was about 25 hours, at the lower end of this range. Since almost all noble gases enter containment quickly after core damage, this may indicate that the flow area to the environment for SGTRs is larger than design leakage. This was tested through one-off calculations and results show that flow rates to environment through the secondary side (combination of MSIV leakage and a stuck open secondary SV) are approximately three orders of magnitude greater than nominal leakage, regardless of any uncertainty sampling. However, since only 10 percent of runs have SGTRs, and because the majority of realizations do not reach rebar failure, design leakage is the highest contributor to release timing uncertainty in regression results for all realizations and for non-SGTR realizations.

The two parameters that show the highest importance in determining whether an SGTR occurs are TUBTHICK, which represents the initial tube thickness of the most damaged steam generator tube in the hottest region, and SVOAFRAC, which represents the fraction of the full open area of a primary or secondary safety relief valve at the time it fails. Both of these parameters have physical bounds, providing high confidence in their uncertainty ranges. The TUBTHICK parameter directly reflects the initial damage state (and effective stress multiplier for creep) of one of the steam generator tubes. The SVOAFRAC parameter influences the depressurization of the RCS after SV failure 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. The magnitude of releases for an SGTR is primarily driven by the time between the SGTR and hot leg creep rupture. The reason is that the SGTR is the primary transport path for radionuclides released from the fuel, until the hot leg ruptures and much of the release is directed back within containment. The average time difference between SGTR and hot leg rupture in these realizations was 28 minutes.

Containment pressure was generally observed to increase from 48 to 72 hours, unless the rebar yield point was reached. This was further investigated by selecting a subset of the single realizations and extending the runs to 72 hours. In the cases where rebar yield was reached, the pressure levels off (to a plateau) and then gradually begins to decrease as the leakage more than compensates for steam generation and heating of the atmosphere. There are marked increases in cesium and iodine environmental release at the point of liner yield, with some increases of an order of magnitude from 48 to 72 hours. Such increases did not occur with rebar yield only. There appear to be no realizations that could ablate through all available concrete by 72 hours, nor are there any BOC realizations that will reach liner yield by 72 hours.

The MACCS results were generated for linear no-threshold (LNT) and two linear with-threshold models (called dose truncation here), referred to as US background and Health Physics Society (HPS) truncation models. The dose truncations (thresholds) are described as: (1) annual dose truncation based on average background plus medical radiation, which is 620 mrem/yr and (2) dose truncation based on the HPS position statement, which states that "the Health Physics Society recommends against quantitative estimation of health risks below an individual dose of 5 rem in one year or a lifetime dose of 10 rem above that received from natural sources."

Like the original SOARCA study, the Surry UA demonstrates that early fatality risks are negligible, essentially zero. LCF risks are even lower than those evaluated in the original SOARCA study. Table 1 shows that the mean LCF risks from this uncertainty analysis, conditional on the occurrence of a STSBO scenario, are below 3×10^{-5}, within 10 miles of the site and the risk diminishes at longer distances. This mean value includes the 10 percent of the realizations with induced SGTRs. For comparison, these values are about a factor of three lower than the SOARCA unmitigated STSBO risks (excluding occurrence of SGTRs) at the same distance ranges (cf., Table 7-4 in NRC, 2013). Furthermore, even the 95th percentile LCF risks from this UA are about a factor of two lower than the mean risks for the unmitigated SOARCA STSBO with induced SGTR (cf., Table 7-6 in 0). This is a meaningful comparison because the top 10 percentile results from this UA represent SGTR realizations; thus, the 95th percentile is approximately the median result for the subset of SGTR realizations.

Table 1. Mean (over weather), individual LCF risks using LNT, conditional on potential unmitigated STSBO occurring (per event) at different radial distances.

	0-10 miles	10-20 miles	0-50 miles
UA Mean	3E-05	1E-05	5E-06
UA Median	6E-06	2E-06	9E-07
UA 5th percentile	7E-07	2E-07	1E-07
UA 95th percentile	2E-04	7E-05	4E-05
SOARCA estimate, STSBO	9E-05	N/A	2E-05
SOARCA estimate, STSBO with SGTR	3E-04	N/A	7E-05

Table 2 provides the regression results for the LNT dose-response for the complete set of regression runs (SGTR and non-SGTR). The top two parameters, TUBTHICK and SVOAFRAC, have an important influence on magnitude and timing of the release and directly influence the evaluation of LCF risk. Both parameters have large values for individual (main) and conjoint contributions. The large conjoint contributions and the lack of significant conjoint influence for the other parameters indicate that the two parameters work together to affect source term. Both parameters are negatively correlated with risk, which indicates that the magnitude of the source term increases as SGT thickness and the open area fraction of a stuck-open safety valve decrease. This combination of parameters makes a SGTR more likely to occur. Additional parameters that significantly increase consequences are time during cycle (fuel burnup, CYCLE), groundshine shielding factor (GSHFAC.2), a parameter controlling leakage from containment (DLEAK), and the cancer fatality risk factor for residual cancers (CFRISK.8), which is the largest single cancer type that contributes to cancer risk. GSHFAC.2 represents the groundshine shielding factor for normal activity. Numbers 1, 2, and 3 are used to represent evacuation, normal activity, and sheltering conditions for GSHFAC. CFRISK.8 represents residual cancer risk, which accounts for all cancers that are not explicitly modeled. Residual cancers are based on doses to the pancreas, which is a surrogate organ to represent generic soft tissues.

Table 2. Mean, individual, LCF risk (LNT dose response) regression results within a 10-mile circular area for all realizations.

	Rank Regression		Quadratic		Recursive Partitioning		MARS		Main Contr.*	Conjoint Contr. *
Final R^2	0.54		0.60		0.86		0.74			
Input	R^2 contr.	SRRC	S_i	T_i	S_i	T_i	S_i	T_i		
TUBTHICK	0.04	-0.20	0.33	0.53	0.30	0.86	0.35	0.80	0.189	0.309
SVOAFRAC	0.03	-0.18	0.23	0.40	0.09	0.55	0.11	0.45	0.082	0.250
CYCLE	0.18	0.44	0.01	0.02	0.01	0.01	0.02	0.02	0.050	0.005
GSHFAC.2	0.13	0.35	0.02	0.05	0.00	0.00	0.01	0.03	0.038	0.011
DLEAK	0.08	0.26	0.01	0.04	0.01	0.01	0.00	0.01	0.022	0.010
CFRISK.8	0.02	0.15	0.02	0.05	0.00	0.06	0.02	0.08	0.011	0.037
SV_STATUS	---	---	0.04	0.04	---	---	---	---	0.006	0.000
DDREFA.8	0.01	-0.12	0.00	0.02	0.00	0.05	0.00	0.03	0.004	0.025
CYSIGA.1	0.02	-0.13	---	---	---	---	---	---	0.004	0.000
TUBETEMP	---	---	0.02	0.02	0.00	0.00	0.01	0.03	0.004	0.006
DEV_DEC_HEAT	0.01	-0.09	0.00	0.03	0.00	0.03	0.01	0.02	0.004	0.015
VDEPOS.1	0.01	0.09	0.01	0.01	0.00	0.04	---	---	0.003	0.011
CFRISK.7	0.01	0.09	---	---	---	---	---	---	0.002	0.000
CFC	0.01	-0.09	---	---	---	---	0.00	0.01	0.002	0.001
CFRISK.6	0.01	0.07	---	---	---	---	0.00	0.02	0.002	0.003
PROTIN.2	---	---	---	---	0.01	0.09	---	---	0.001	0.023
CHEMFORMCS	0.01	-0.06	---	---	---	---	---	---	0.001	0.000
SGTRLOC	0.00	0.06	0.00	0.01	---	---	---	---	0.001	0.002
CFRISK.2	---	---	0.00	0.03	---	---	---	---	0.001	0.005
LA.140_ICH.9	---	---	0.00	0.04	0.00	0.03	0.00	0.01	0.000	0.018
PARTSHAPE	---	---	0.00	0.01	0.00	0.01	---	---	0.000	0.004
CHEMFORMI2	---	---	---	---	0.00	0.02	0.00	0.01	0.000	0.009

** highlighted if main contribution larger than 0.02 or conjoint contribution larger than 0.1*

All consequence results are presented as conditional risks, which are the risks conditional on the accident occurring. The emergency phase used in this analysis is the first seven days following the beginning of release to the environment. The long-term phase immediately follows the emergency phase and lasts for 50 years. Results for the LNT dose-response show the large majority of the LCF risk is from the long-term phase at all five distance intervals evaluated, even for the realizations with SGTR. The mean values of the fraction of risk from the emergency phase are 1 percent within 10 miles and about 15 percent at distances beyond 10 miles. Only a handful of realizations have emergency-phase contributions to risk that exceed those from the long-term phase.

The results for the distance from 0 to 10 miles have a very different character than those for the other distance ranges, because evacuation is very effective in reducing risk during the emergency phase for the population living within the emergency planning zone (EPZ). Most of the overall risk within 10 miles is to the 0.5 percent of the public that is assumed to not evacuate; however, some of the risk is to the slowly evacuating cohorts for the realizations with relatively early releases, mainly the realizations with SGTR.

A set of sensitivity analyses was also completed with this project. Some of the sensitivities were performed to support parameter development and more detailed sensitivities were performed to better understand the contributions of individual uncertain inputs to the uncertainty in analysis results. The following MELCOR sensitivity analyses were performed to provide targeted insights into the results of the analysis.

4 SENSITIVITY ANALYSES

4.1 *Sensitivity analysis on concrete type*

The Surry SOARCA analyses (NRC, 2012; NRC, 2013) used limestone aggregate in the calculation, which was the aggregate identified in the reference Modular Accident Analysis Program (MAAP) file. However, during the investigations into this UA, it was discovered that Surry actually has a basaltic aggregate. A sensitivity analysis was conducted to understand the effects of the different concrete types. The basaltic concrete produced a more vigorous molten core-concrete interaction (MCCI) than the limestone concrete; however, the limestone concrete showed approximately 30 percent greater release of non-condensable gases (NCGs). There was little difference in containment pressure and temperature between the calculations even though the mass of gas generated by core concrete interaction varied by approximately 30 percent. The key observation was that most pressure in containment is attributable to the partial pressure of steam. The partial pressures of NCGs generated by MCCI do not contribute much to the overall pressure. The environmental release fraction from limestone concrete for cesium and iodine was found to be greater (~25% greater for cesium and ~50% greater for iodine) than from basaltic concrete at the end of the 48 hour simulation. This was attributed to the increased containment pressurization in the limestone concrete analysis which reached liner yield (i.e., functional failure of containment) at approximately 36.5 hours, compared to the basaltic concrete analysis which reached liner yield at approximately 41 hours.

4.2 *Joint sensitivity analysis on number of steam generator tubes ruptured*

A joint sensitivity analysis was conducted with sampling performed on number of tubes, primary and secondary SV open fraction, and SGTR location to determine the effect of more than one SG tube failing on cesium and iodine environmental release fractions. Other uncertain inputs were fixed at their base case values. In the joint sensitivity analysis results, releases for one tube were slightly higher than the SGTR realizations observed in the full UA. There were step increases in release fractions for the failure of two and three tubes, but no significant difference from three to five tubes. This indicates that when three or more tubes fail, flow to environment is limited by main steam isolation valve (MSIV) leakage and the stuck open secondary SV, and not by the flow through the SG tubes. Another key result is that releases from one tube increase more gradually while three to five tubes has more of a quick puff release. The number of tubes parameter dominated results, with none of the other three sampled parameters having a significant additional impact. A large factor in the total release fractions is the time of hot leg creep relative to the SGTR. This timing is primarily due to depressurization rate, based on the number of tubes failing and exacerbated by the primary SV open fraction. Out of 97 Monte Carlo realizations in this joint sensitivity, one had sampled five tubes failing and had no hot leg creep rupture within the 48 hour simulation time, and this realization had the highest release fractions.

4.3 *MACCS sensitivity analyses*

Four MACCS sensitivity analyses were conducted using one of the larger source terms from MELCOR. The sensitivities evaluated emergency phase durations of 15 and 30 days, intermediate phase duration of 6 months, and a 4-day (instead of 7-day) dose projection period for the emergency phase. The results were all essentially the same with one exception, the risks for the 0- to 10-mile distance interval was noticeably larger for the case when the intermediate phase is 6 months than when there was no intermediate phase. The increase in risk for the 0- to 10-mile interval indicates that less decontamination occurs when the intermediate phase is included and that more individuals receive a larger dose when they return home than receive a smaller dose. This can occur when the 6 months of decay and weathering provided by the intermediate phase brings the dose levels below the habitability threshold without the need to decontaminate. However, these dose levels can be higher than they would have been if decontamination were performed because the MACCS decontamination modeling reduces dose levels by pre-defined fixed factors of 3, or 15 if a reduction factor of 3 is insufficient to restore habitability.

The habitability criteria applied in these sensitivity analyses is considered to be an important uncertain parameter, but was not evaluated with this Surry UA because a detailed sensitivity analysis was performed with the Peach Bottom UA (NRC, 2016). The Peach Bottom analysis showed, as would be expected, that when the dose truncation models were used, the LCF risks within the EPZ were orders of magnitude lower when the habitability criteria was below the dose truncation level. Beyond the EPZ, the habitability criteria showed a smaller effect on the overall LCF risk when a dose truncation model was applied.

5 SUMMARY

As described above, this Surry UA modeled distributions for parameter values that historically were modeled with fixed values and applied multiple regression techniques to support an understanding of the results. Such an analysis produces substantial information which is described in detail in the NRC's draft SOARCA Surry UA report. A summary of important insights is provided below:

- SGTRs occurred in about 10 percent of the Monte Carlo realizations and had release fractions one to two orders of magnitude larger.
- SGTRs are induced by both a thermal and pressure element.
- In the number of SG tubes joint sensitivity analysis, one realization with 5 tubes failing had no hot leg creep rupture and this led to the highest release fractions.
- In most of the Monte Carlo realizations, iodine and cesium environmental release fractions were higher early in the transient than the Surry SOARCA calculation (NRC, 2013), but all were significantly 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 (gradually degrading containment versus sudden catastrophic failure). All of these, on average, 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 Surry SOARCA calculation 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 mean population-weighted 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 uncertainty analysis (NRC, 2016). This is true when a single dose-response model (LNT) is used, but uncertainties in risks created by uncertainties in dose-response model are large and most likely would have altered this conclusion if dose response had been included as part of the integrated uncertainty analysis.

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

- For early hydrogen production primary system depressurization followed by time at cycle, were most important, but the total magnitude of hydrogen was most influenced by the effective fuel melt temperature.
- For non-SGTR realizations (90% of the realizations), for Cs and I release fractions, the most important parameters were time at cycle, design leakage, the containment failure curve, and the dynamic shape factor. For iodine, the amount that was assumed to be gaseous was very important, determined by time at 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 Cs and I release fractions for SGTR releases.
- For LCF risk, TUBTHICK and SVOAFRAC are the most influential parameters. These two parameters largely determine whether the accident progresses toward an SGTR. Thus, they have an important influence on magnitude and timing of the release and directly influence LCF risk.
- In addition to TUBTHICK and SVOAFRAC, three other parameters have a significant effect on LCF risk. These are time at cycle (fuel burnup), groundshine shielding factor, and design leakage from the containment. Time at cycle affects both the amount of decay heat in MELCOR and fission product inventory in MACCS. Both of these increase LCF risk as time at cycle increases. Decay heat quickly approaches steady state early in the fuel cycle while some influential fission products, like Cs-137, increase approximately linearly during the fuel cycle. Groundshine shielding factor directly influences dose through the groundshine pathway, which is the dominant dose pathway during the long-term phase. Finally, design leakage influences the release of radioactivity through the containment, which is especially important when a SGTR does not occur.
- The most important parameter in the Peach Bottom UA, 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 made narrower in 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.

Similar to the Peach Bottom UA (NRC, 2016), the results of this Surry UA corroborate the following conclusions from the SOARCA project (NRC, 2012):

- Latent cancer and early fatality risks from severe nuclear accident scenarios modeled are smaller than those projected in NUREG/CR-2239.
- The delay in releases calculated provides more time for emergency response actions (such as evacuating or sheltering).
- "Essentially zero" absolute early fatality risk is projected for the potential accidents studied.

Insights from the Surry UA project will be used to help identify key sources of uncertainty for the NRC's Level 3 PRA project, and to help disposition post-Fukushima regulatory actions and knowledge management activities.

REFERENCES

NRC, 2012. *NUREG-1935, State-of-the-Art Reactor Consequence Analyses (SOARCA) Report*. Washington, DC: U.S. Nuclear Regulatory Commission.

NRC, 2016. *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*. Washington, DC: U.S. Nuclear Regulatory Commission.

NRC, 2012a. *SECY-12-0092, "State-of-the-Art Reactor Consequence Analyses – Recommendation for Limited Additional Analysis*. Washington, DC: U.S. Nuclear Regulatory Commission.

NRC, 2013. *NUREG/CR-7110 Volume 2, Rev. 1, "State-of-the-Art Reactor Consequence Analysis Project Volume 2: Surry Integrated Analysis*. Washington, DC: U.S. Nuclear Regulatory Commission.