State-of-the-Art Reactor Consequence Analyses Project: Sequoyah Uncertainty Analysis Methods and Insights

S. Tina Ghosh[a], Doug Osborn[b], Nathan Bixler[b], Kyle Ross[b], Dusty Brooks[b]
[a] U.S. Nuclear Regulatory Commission, Washington, DC, USA
[b] Sandia National Laboratories, Albuquerque, NM, USA

Abstract: Through the application of modern analysis tools and techniques, the state-of-the-art reactor consequence analyses (SOARCA) project developed a body of knowledge regarding the realistic outcomes of potential severe accidents at nuclear power reactors. In NUREG-1935, "State-of-the-Art Reactor Consequence Analyses (SOARCA) Report", SOARCA analyses of the Peach Bottom Atomic Power Station and Surry Power Station pilot plants revealed insights into the accident progression for important scenarios in a boiling water reactor (BWR) with a Mark I containment design and a pressurized water reactor (PWR) with a large dry (subatmospheric) containment design as well as the offsite consequences of potential radioactive releases. This paper discusses the subsequent analysis of station blackouts (SBOs) at the third SOARCA pilot plant, Sequoyah Nuclear Plant, which expanded on the SOARCA body of knowledge. Hydrogen combustion has long been known to be a potential challenge to the ice condenser containment. The Sequoyah SOARCA analysis examines phenomenology and modeling unique to the ice condenser design including the behavior of hydrogen and the potential for early containment failure from an energetic hydrogen combustion. The SOARCA Sequoyah analysis confirmed insights from previous analyses, yielded new insights, and contributed to the U.S. Nuclear Regulatory Commission's disposition of post-Fukushima-accident regulatory initiatives for ice condenser containments.

Keywords: SOARCA, Sequoyah, uncertainty analysis, severe accident analysis, MELCOR, MACCS.

1. INTRODUCTION

Through the application of modern analysis tools and techniques, the state-of-the-art reactor consequence analyses (SOARCA) project developed a body of knowledge regarding the realistic outcomes of potential severe accidents at nuclear power reactors. In NUREG-1935, "State-of-the-Art Reactor Consequence Analyses (SOARCA) Report" [1], SOARCA analyses of the Peach Bottom Atomic Power Station and Surry Power Station pilot plants revealed insights into the accident progression for important scenarios in a boiling water reactor (BWR) with a Mark I containment design and a pressurized water reactor (PWR) with a large dry (subatmospheric) containment design as well as the offsite consequences of potential radioactive releases. This paper discusses the subsequent analysis of station blackouts (SBOs) at the third SOARCA pilot plant, Sequoyah Nuclear Plant, which expanded on the SOARCA body of knowledge for the next most prevalent containment design in the U.S., a PWR with an ice condenser containment. Compared to a PWR large dry containment, an ice condenser containment design has smaller volume and has a lower design pressure; therefore, the containment cannot absorb as much energy from a deflagration despite the presence of ice. Hydrogen combustion has long been known to be a potential challenge to the ice condenser containment. The Sequoyah SOARCA analysis examines phenomenology and modeling unique to the ice condenser design including the behavior of hydrogen and the potential for early containment failure from an energetic hydrogen combustion. The SOARCA Sequoyah analysis confirmed insights from previous analyses, yielded new insights, and contributed to the U.S. Nuclear Regulatory Commission's disposition of post-Fukushima-accident regulatory initiatives for ice condenser containments.

The Sequoyah SOARCA integrated uncertainty analysis (UA) focused on an unmitigated short-term station blackout scenario involving an immediate loss of offsite and onsite AC power. The Sequoyah

SOARCA analyses were performed primarily with two computer codes, MELCOR for severe accident progression and the MELCOR Accident Consequence Code System (MACCS) for offsite consequences, which are the same tools used in previous SOARCA efforts. The UA input parameter selections included 600 samples resulting in 567 successful MELCOR Monte Carlo realizations. Each MELCOR source term was coupled with a unique input parameter set for the 567 MACCS UA realizations.

These STSBO UA results were examined in detail using both quantitative and qualitative approaches. Selected realizations from the UA were identified and examined to qualitatively evaluate characteristics important to accident progression and offsite consequences. Four regression techniques were used to identify input parameters contributing to key accident progression characteristics and public health impacts. Measures of the main (individual, independent) contribution of the uncertain parameter on the result metric and the conjoint influence (influence of two or more input parameters acting together) of the parameter on the result metric were determined. These two results were calculated as weighted averages of the overall contributions from the four regression techniques. In addition, a stability analysis was performed for the result metrics of interest (e.g., cesium and iodine release to the environment) using a bootstrapping method, to gain an understanding of the level of convergence in the statistical results.

2. SCOPE AND APPROACH

While the Peach Bottom and Surry SOARCA analyses in NUREG-1935 considered a variety of scenarios including short-term and long-term SBOs, loss of vital AC power equipment for Peach Bottom, steam generator tube rupture (SGTR) and interfacing systems loss of cooling accidents (i.e., a high-pressure system causing a connected lower-pressure system to fail) for Surry, the Sequoyah SOARCA analysis focused specifically on short-term SBO (STSBO) and long-term SBO (LTSBO) scenarios only [2]. These SBO scenarios involve an immediate loss of offsite and onsite AC power. In the STSBO variation, early failure of the turbine driven auxiliary feedwater (TDAFW) system (a backup heat removal system independent of AC power) is assumed and direct current (DC) (battery) power is also immediately unavailable; thus, the accident can progress to core damage within the "short term" (generally a few hours). In the LTSBO variation, DC power is assumed to be available until station batteries deplete and the TDAFW system is initially available; thus, absent any effective intervention, the accident would progress to core damage within the "long term" (generally beyond a day). These are important scenarios for all light-water reactors in general, and for an ice condenser plant because of its reliance on AC-powered igniters for hydrogen control in the event that the accident progresses to core damage with associated hydrogen production. The possibility of an SBO-induced SGTR is not modeled in this analysis since the focus here is on containment structural performance, and the Surry SOARCA uncertainty analysis included a more detailed treatment of possible steam generator tube failures.

NRC and Sandia National Laboratories (SNL) staff used updated and benchmarked standardized plant analysis risk (SPAR) models and available plant-specific external events information to identify the SBO scenario variations for this analysis. SBO scenarios can be initiated by external events such as a fire, flood, or earthquake. The Sequoyah SOARCA analysis assumes that the SBO is initiated by a low probability severe seismic event because this is a challenging case in terms of timing, equipment failure, and evacuation of the areas around the plant. The contribution to core damage frequency for the LTSBO was estimated at one event per approximately 100,000 years of reactor operation (~1E-5 per reactor operating year). The contribution to core damage frequency for the STSBO is estimated at one event per approximately 500,000 years of reactor operation (~2E-6 per reactor operating year). No new work on estimating SBO frequencies was undertaken for this study. The estimated frequency information is provided only to help place this consequence study in context.

The Sequoyah SOARCA analyses were performed primarily with two computer codes, MELCOR for severe accident progression and the MELCOR Accident Consequence Code System (MACCS) for offsite radiological consequences, which are the same tools used in previous SOARCA efforts. During the analysis, improvements to the accident progression and consequence analysis computer models and tools were incorporated to provide the current state-of-the-art severe accident modeling practices. MELCOR models the following:

- Thermal-hydraulic response in the reactor coolant system, reactor cavity (below the reactor vessel), containment, and confinement buildings (e.g., shield building);
- Core heatup, degradation (including fuel cladding oxidation, hydrogen production, and fuel melting), and relocation;
- Core-concrete interaction in the cavity after lower reactor vessel head failure;
- Hydrogen production, transport, combustion, and mitigation; and
- Fission product transport and release to the environment.

MACCS models the following:

- Atmospheric transport and deposition of radionuclides released to the environment;
- Emergency response and long-term protective actions;
- Exposure pathways;
- Acute and long-term doses to a set of tissues and organs; and
- Early and latent health effects for the affected population resulting from the doses[*].

The Peach Bottom and Surry SOARCA studies (NUREG/CR-7110 Volume 1 [4] and Volume 2, respectively [5]) were comprised of deterministic analyses using point estimates for input parameter values, followed by uncertainty analyses (Peach Bottom Uncertainty Analysis documented in NUREG/CR-7155 [6] and DRAFT Surry Uncertainty Analysis [3], respectively), which sampled distributions representing input uncertainty to generate multiple results to represent a range of potential outcomes. The Sequoyah SOARCA analysis integrates probabilistic consideration of uncertainty into accident progression and offsite consequence analyses in parallel with deterministic calculations that are presented within this study.

Because this study had a particular focus on the potential for early containment failure, and ice condenser containment-specific issues, an uncertainty analysis (UA) was included for the scenario with the greatest potential for early containment failure: the unmitigated STSBO[†] scenario. This scenario was an evaluation of an unmitigated STSBO without hydrogen igniters, and without the presence of random ignition sources[‡]. The UA included three active ignition sources:

1) ignition from hot gases exiting the hot leg failure location,
2) ignition from hot gases exiting the pressurizer relief tank (PRT), and
3) the ex-vessel debris following the reactor pressure vessel (RPV) melt-through.

Key MELCOR (accident progression) and MACCS (offsite consequence) input parameters were selected and assigned distributions that account for parameter uncertainty. The UA input parameter uncertainty treatment included 600 Monte Carlo samples resulting in 567 successful MELCOR realizations (or outcomes). Each MELCOR source term was coupled with a unique input parameter set for the 567 MACCS UA realizations.

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[*] MACCS also models economic and societal consequences such as the population subject to protective actions, however, these were not used in the SOARCA project.

[†] Because the unmitigated STSBO does not credit human actions, the UA does not address human actions.

[‡] An initial draft of the Sequoyah SOARCA analysis included two variations of the unmitigated STSBO – with and without random ignition. The 'with' random ignition UA was not conducted for this final report. The reader is directed to Section 4 for further discussion and the *DRAFT* report of this effort: https://www.nrc.gov/docs/ML1609/ML16096A374.pdf.

These STSBO UA results were examined in detail using both quantitative and qualitative approaches. Selected realizations from the UA were identified and examined to qualitatively evaluate characteristics important to accident progression and offsite consequences. Four regression techniques were used to identify input parameters contributing to key accident progression characteristics and public health impacts. Measures of the main (individual, independent) contribution of each uncertain parameter on the result metric and the conjoint influence[§] of the parameter on the result metric were determined. These two results were calculated as weighted averages of the overall contributions from the four regression techniques. In addition, a stability analysis was performed for the result metrics of interest (e.g., cesium and iodine release to the environment) using a bootstrapping method, to gain an understanding of the level of convergence in the statistical results.

The focus of this UA was on epistemic parameter uncertainty. NRC's guidance[**] on the treatment of uncertainties notes two other types of epistemic uncertainty; completeness uncertainty (i.e., missing scope) and model uncertainty (i.e., alternate conceptual models). Completeness and model uncertainties were outside the scope of this UA. A systematic evaluation of model uncertainties could reveal additional important uncertainties that were not captured within the scope of this UA.

Although a systematic evaluation of model uncertainty was not within the scope of this UA, separate sensitivity analyses were conducted to understand the variation in results arising from some alternative modeling approaches. For accident progression, sensitivity analyses were conducted to examine the effect of including hydrogen igniters and the effect of reactor coolant pump (RCP) seal leakage. For offsite consequences, sensitivity analyses were conducted to examine the effect of delays in evacuation due to infrastructure damage, the reduced protection offered by degraded infrastructure caused by the seismic event, the effect of using a single weather year for the analysis, and the potential influence of the dose-response model on cancer risk.

For the MELCOR accident progression analysis, insights from the SOARCA DRAFT Surry UA [3] were leveraged to identify a reduced set of parameters to include in the Sequoyah integrated UA. Because of the Sequoyah SOARCA analysis' focus on insights unique to the ice condenser containment and potential vulnerabilities to hydrogen challenges, ice condenser containment-specific and hydrogen-specific considerations added new parameters to the list. Table 1 lists the 13 MELCOR input parameters varied as part of the unmitigated STSBO UA accident progression.

Table 1. Uncertain MELCOR parameters used in the unmitigated STSBO UA

Sequence Related Parameters
Primary safety valve stochastic number of cycles until failure-to-close
Primary safety valve open area fraction after failure
Secondary safety valve stochastic number of cycles until failure-to-close
Secondary safety valve open area fraction after failure
In-Vessel Accident Progression
Melting temperature of the eutectic formed from fuel and zirconium oxides
Oxidation kinetics model
Ex-Vessel Accident Progression

§ Conjoint influence is the influence of two or more input parameters acting together, which may have synergistic effects that would not be determined by studying the influence of each parameter separately and individually.

** See for example, NRC Regulatory Guide 1.174: https://www.nrc.gov/docs/ML1009/ML100910006.pdf. NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision making," contains some useful guidance on how to treat model uncertainties. While this guidance is geared toward risk-informed license applications, the general principles apply to any analysis.

Lower flammability limit hydrogen ignition criterion for an ignition source in lower containment
Containment rupture pressure
Barrier seal open area
Barrier seal failure pressure
Ice chest door open fraction
Particle dynamic shape factor
Time within the Fuel Cycle
Time-in-cycle[††]

For the offsite consequence analysis, parameters varied as part of the STSBO UA are shown in Table ES-2. These uncertain parameters are the same as those in the SOARCA *DRAFT* Surry UA [3] with two new parameters added: (1) a time-based crosswind dispersion coefficient and (2) a parameter related to weather-forecasting-time used in the keyhole evacuation model. One parameter that was made uncertain in the previous UAs, cloudshine shielding factor, was fixed as a point value in this analysis because cloudshine contributes very little to the overall risk. The results are presented as conditional individual latent cancer fatality (LCF) risk and conditional individual early fatality (EF) risk and are statistically averaged (reported as the mean) over weather conditions.

The Sequoyah MACCS model was developed assuming the large seismic initiating event that disrupts the Sequoyah plant systems also affects the evacuation routes. This Sequoyah SOARCA approach differs from earlier Peach Bottom and Surry 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 within the 10-mile emergency planning zone (EPZ) are unusable. The infrastructure beyond the Sequoyah EPZ is assumed to be unaffected by the earthquake. It is difficult to consider all potential scenarios with respect to damage incurred within the EPZ from a large earthquake, such as the conditions of individual houses or buildings, the damage to roads in addition to bridges, the ability to evacuate or to shelter, etc. Therefore, the primary factors modeled in this study are the evacuation speeds and delays due to the loss of roadways with bridges. Specifically, the loss of roadways to exit the EPZ is expected to result in delays to find alternate routes and in decreased evacuation speeds due to increased traffic congestion on a suboptimal road network. The evacuation delays and decreased travel speeds are evaluated to encompass other factors affecting the ability to evacuate.

Table ES-2 Uncertain MACCS parameter groups used in the unmitigated STSBO UA

Epistemic Uncertainty
Dispersion
Crosswind Dispersion Linear Coefficient
Vertical Dispersion Linear Coefficient
Time-Based Crosswind Dispersion Coefficient
Deposition
Wet Deposition Coefficient
Dry Deposition Velocities
Emergency Response
Keyhole Weather Forecast
Evacuation Delay

[††] Three points in the fuel burn-up cycle were sampled to represent beginning-of-cycle (BOC), middle-of-cycle (MOC), or end-of-cycle (EOC).

Epistemic Uncertainty
Evacuation Speed
Hotspot Relocation Time
Normal Relocation Time
Hotspot Relocation Dose
Normal Relocation Dose
Shielding Factors
Groundshine Shielding Factors
Inhalation Protection Factors
Early Health Effects
Early Health Effects LD_{50} Parameter
Early Health Effects Exponential Parameter
Early Health Effects Threshold Dose
Latent Health Effects
Dose and Dose Rate Effectiveness Factor
Lifetime Cancer Fatality Risk Factors
Long-Term Inhalation Dose Coefficients
Aleatory Uncertainty
Weather

The Sequoyah UA follows the approach developed for the previous Peach Bottom [4] and draft Surry [3] UAs. Lessons learned from the previous UAs and feedback from the NRC's Advisory Committee on Reactor Safeguards (ACRS) were considered, as well as additional knowledge gained since the Peach Bottom and Surry best estimate calculations [5,6]. Objectives for this work include: developing insights into the overall sensitivity of results to uncertainty in selected modeling inputs; evaluating the likelihood of early containment failure and resulting consequences; identifying the most influential input parameters contributing to accident progression and offsite consequences through application of a set of regression analyses; informing the NRC's Site Level 3 Probabilistic Risk Assessment (PRA) project and post-Fukushima-accident regulatory activities.

3. RESULTS AND INSIGHTS

A primary goal of the Sequoyah UA was to investigate the potential for an ice condenser containment to fail early from a hydrogen deflagration in a severe accident situation. The containment end-state results are accordingly characterized into three general outcomes:

1. Late containment failure due to a slow pressurization of the containment from the core-concrete interaction and steam
    - The most common outcome in the 567 Monte Carlo realizations

2. No containment failure within 72 hours of the onset of the STSBO
    - Almost exclusively beginning-of-cycle (reflecting an accident that occurs soon after refueling) sampled realizations with lower decay heat power

3. Early containment failure due to combustion of hydrogen generated in-vessel, following the first ignition
    - Only a few realizations with this outcome.

The behavior of the safety valves is very important in affecting the accident progression; most notably in-vessel hydrogen production and release to the containment, and the potential for early containment failure.

A key insight from the STSBO uncertainty analysis is that early containment failure can occur only when a primary side safety valve fails-to-close, and the safety valve failure-to-close probability remains highly uncertain based on current state-of-knowledge. While there is some operational data on safety valve failure-to-close (i.e., main steam safety valves following scram events), the data is not under the exact conditions that the safety valves would experience in the STSBO scenario. Safety valve performance is clearly important in an STSBO, yet the basis for a more confident modeling of the uncertainties in safety valve behavior is currently lacking. The analysis here attempts to address the lack of applicable data (with respect to both failure rates and failure mechanisms) on safety valve failure-to-close in one reasonable way that has been positively reviewed by experts outside the team. However, currently there is no established expert consensus on the best way to model safety valve failure-to-close.

The safety valve modeling was updated over the course of this project, because of its importance in affecting the accident progression. A draft version of this analysis was prepared using different assumptions for the primary safety valve (SV) failure attributes. The very small number of early containment failures within this UA compared to this draft are due primarily to changes in the primary SV failure attributes. After the draft UA was completed, the SV failure attributes were changed based on discussions with nuclear valve testing personnel and closer examination of Licensee Event Reports. In particular, the SV behavior was updated from the draft UA to reflect the following expert judgments based on the additional information gathered:

- If an SV is going to fail to close (FTC), then it will most likely do so on initial demand,
- If an SV functions per design on initial demand, then it will most likely function on all subsequent demands,
- If an SV experiences an FTC, then it will likely be in either a weeping (mostly closed) or mostly open position,
- An SV is very unlikely to fail to open when demanded to do so (i.e., fail fully closed),
- Passing hot liquid is not necessarily threatening to an SV but passing cold fluid is, and
    - Cold being relative to the valve's design conditions
- While there are differences between the main steam and reactor coolant system (RCS) SVs, it was judged defensible to apply the main steam SV operational data to both the main steam and RCS SVs too due to a lack of operational data for RCS SVs.

The amount of hydrogen produced in-vessel is generally less when the pressurizer SV continues cycling until hot leg failure with the primary system at high pressure. For these high pressure cases, the heat from the core is more efficiently transferred from the vessel to the reactor coolant system piping. The combination of a high differential pressure across the hot leg to the containment, and the more effective heating at high pressure results in hot leg failure earlier in the core degradation progression. The first ignition source is the failure of the hot leg, which is associated with less in-vessel hydrogen generation, and a lower amount of hydrogen transported to the containment. The early deflagration from the smaller amounts of hydrogen in the high pressure cases did not fail the containment. Furthermore, the subsequent deflagrations also did not challenge the containment integrity because they occurred closer to the lower flammability limit of hydrogen due to the presence of active ignition sources (e.g., hot gases from the primary system or ex-vessel debris).

If containment did not rupture early (within 12 hours), then the subsequent hydrogen burns were not energetic enough to rupture the steel containment vessel later in the sequence. The late burns (after 12 hours) are less energetic due to frequent burning of smaller quantities of combustible gases near the lower flammability limit (i.e., ignited by aerosols and hot gases from ex-vessel CCI). As the burns consume oxygen, the oxygen concentration in containment eventually decreases to the point where it is insufficient to support further burning. Although the deflagrations cease, the containment continues to pressurize and heat up from the ex-vessel CCI non-condensable gas generation and the resulting vaporization of water from the melted ice. The pressurization is monotonic and most often pressurizes containment to rupture prior to 72 hours (end of simulation time). None of the BOC realizations overpressurized containment by 72 hours. Figure 1 shows the containment pressure response for the STBO UA realizations, color coded by time-in-cycle. Note that the abrupt late depressurizations in the range of about 70 to 90 psia are associated with containment failure.

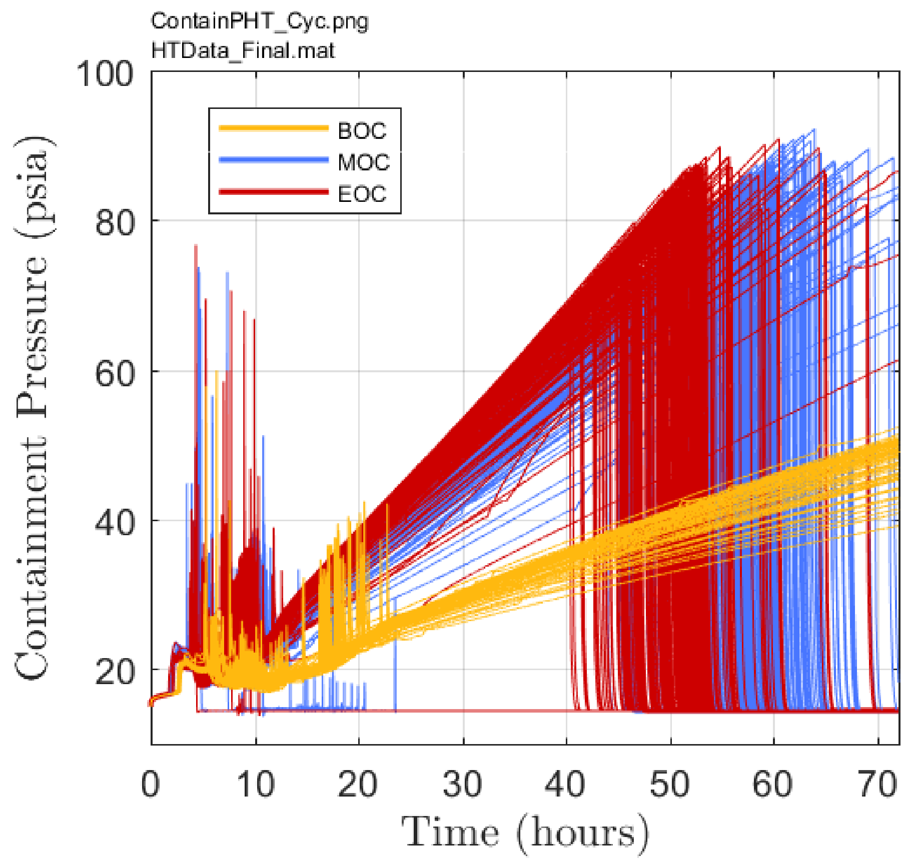

Figure 1. Containment pressure response for the STSBO UA realizations

Consistent with the insights described above and past studies (such as the draft Surry SOARCA UA), regression analyses indicate that the most influential uncertain parameters on in-vessel hydrogen production were the oxidation model, time in burnup cycle when accident occurs, aggregate primary safety valve cycles, and U-Zr-O eutectic melt temperature.

For both cesium and iodine environmental release (Figure 2), regression analyses indicate the time-in-cycle is significant; Realization 266 (Rlz 266) is the reference realization because its results are close to the population medians for figures-of-merit. Releases vary from ‘nearly zero’ (at 72 hours) for BOC realizations, then increase in magnitude and occur earlier with increasing burnup for middle-of-cycle (MOC) and then end-of-cycle (EOC) realizations. The aggregate number of primary SV cycles experienced (priSVcycles, due to SV FTC or RCS depressurization by other means) is also significant, and the results are consistent with deterministic analyses. The primary SV cycles parameter also has high interaction effects (identified from non-linear regression techniques) for both the cesium and iodine environmental release. With regard to the magnitude of the environmental release, no other parameters were identified as having significant effects on cesium or iodine release to the environment. In terms of significant effects only in interaction with other parameters, the U-Zr-O eutectic melt temperature was identified as significant for both cesium and iodine environmental releases, and the containment rupture pressure was identified as significant for only the iodine environmental release. Generally, cesium and iodine environmental releases are minimal until about 42 hours into the simulation, and increase significantly from 48 hours to the end of the simulation (72 hours).

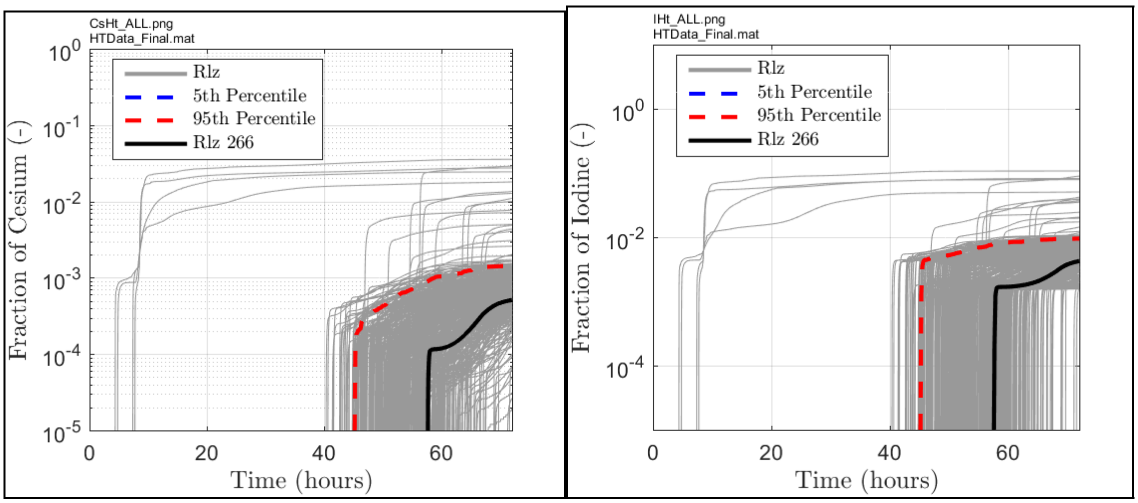

Figure 2. Cesium (left) and iodine (right) environmental release fraction horsetails based on the STSBO UA realizations.

The MACCS offsite consequence analysis results are discussed in a companion paper at this PSAM 14 conference.

4. CONCLUSIONS

The Sequoyah SOARCA analyses provide valuable insights on potential accident progression and consequences for an unmitigated station blackout severe accident at the Sequoyah Nuclear Generating Station. An important focus of the analyses was investigating the susceptibility of the ice condenser containment to rupture from a hydrogen deflagration. The analyses suggest that rupturing of an ice condenser containment by a hydrogen deflagration, while possible, is unlikely. The analyses also suggest that in the severe accident scenario modeled, a safety valve on the primary side (pressurizer) of the reactor coolant system would need to fail to close for an ice condenser containment to rupture from a hydrogen deflagration. A containment rupture from a hydrogen deflagration would come early (i.e., within hours of the loss of all electrical power for an STSBO). Considerably more likely than an early containment rupture is a late rupture (i.e., days after the loss of all electrical power) due to gradual over-pressurization from fission product decay incessantly driving steam production and core-concrete interaction. The relative likelihood of different containment failure times in the modeled STSBO scenario is highly influenced by uncertainty in the primary safety valve failure-to-close parameters, and considerable uncertainty remains in the distributions of these key safety valve parameters in the current state-of-knowledge. Sensitivity calculations conducted as part of this study reinforce the results of past analyses (e.g., NUREG-1150) 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 EPZ), resulting individual LCF risks are small and individual EF risks are essentially zero. Offsite consequence results are discussed in more detail in a companion paper to this conference.

Sequoyah SOARCA insights, while specific to Sequoyah, can be used to obtain insights for other PWRs with ice condenser containments. However, additional work would be needed to assess the impact of differences in plant-specific designs and site-specific characteristics.

The SOARCA Sequoyah analysis confirmed insights from previous analyses, e.g., that early containment failure is possible even if unlikely, and that igniters can avert early containment failure; yielded new insights, such as the nature of the influence of safety valve behavior on potential early containment; and contributed to the U.S. NRC's disposition of post-Fukushima-accident regulatory initiatives for ice condenser containments.

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References

[1] U.S. NRC, "State-of-the-Art Reactor Consequence Analyses Project," NUREG-1935, U.S. Nuclear Regulatory Commission, Washington, DC, USA, 2012.

[2] Sandia National Laboratories, "State-of-the-Art Reactor Consequence Analyses Project: Sequoyah Integrated Deterministic and Uncertainty Analyses," NUREG/CR-7245, U.S. Nuclear Regulatory Commission, Washington, DC, USA, 2018.

[3] K. Ross, N. Bixler, S. Weber, C. Sallaberry, and J. Jones, "State-of-the-Art Reactor Consequence Analysis Project: Uncertainty Analysis of the Unmitigated Short-Term Station Blackout of the Surry Power Station," ADAMS Accession Number ML15224A001, U.S. Nuclear Regulatory Commission, Washington, DC, USA, 2015.

[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, Washington, DC, USA, 2016.

[5] N. Bixler, R. Gauntt, J. Jones, and M. Leonard, "State-of-the-Art Reactor Consequence Analyses Project, Volume 1: Peach Bottom Integrated Analysis," NUREG/CR-7110, Vol. 1, Rev. 1, U.S. Nuclear Regulatory Commission, Washington, DC, USA, May 2013.

[6] Sandia National Laboratories, "State-of-the-Art Reactor Consequence Analyses Project, Volume 2: Surry Integrated Analysis," NUREG/CR-7110, Vol. 2, Rev. 1, U.S. Nuclear Regulatory Commission, Washington, DC, USA, August 2013.