Peer Review Comment Resolution Report

April 30 - December 3, 2010

The purpose of this report is to provide the Peer Review Committee with the State-of-the-Art Reactor Consequence Analysis (SOARCA) team's resolution of each peer review comment related to the best estimate analysis and received between April 30 and December 3, 2010. In some cases, previous comments (received before the April 2010 draft peer review report) were included to provide a complete understanding of the entire comment. This report is organized by individual peer reviewers (see the Contents below). The source document and its associated page number are identified for each comment. Comments are extracted directly from the source documents with no changes. The SOARCA report sections that are identified in the resolutions refer to the revised reports that will be provided to the Peer Review Committee before the final meeting on the best estimate analysis.

Contents

Note: Mr. Roger Kowieski indicated in the April 30, 2010 peer review draft report and the December 3, 2010, peer review memorandum that all of his comments had already been adequately addressed.

KEN CANAVAN

April 30, 2010 SOARCA Peer Review Draft Report

1. **Page 10**

Comment:
There is a possibility that certain accident sequences, while not-dominant, may have increase risk in terms of increased consequences. While these sequences may not dominant the risk, in terms of either frequency and/or consequence, they could be contributors. Collections of several lower order sequences could have higher consequence than SOARCA evaluated and could also contribute. While SOARCA did indeed capture the most likely sequences and did accurately capture the consequence from these sequences.

Resolution:
As explained in Section 2.1, “Approach,” of the Main Report, the State-of-the-Art Reactor Consequence Analyses (SOARCA) team considered and selected accident scenarios (sequence groups, rather than individual sequences) based on both likelihood and potential consequences. Core-damage sequences from previous staff and licensee probabilistic risk assessments (PRAs) were identified and binned into core-damage groups. A core-damage group consists of core-damage sequences that have similar timing for important severe accident phenomena and similar containment or engineered safety feature operability. The *groups* (not individual sequences) were screened according to their approximate core-damage frequencies to identify the most significant groups. Since core-damage groups, i.e., scenarios, were considered, many individual lower-order sequences would be captured in the aggregation into groups.

The SOARCA team used selection criteria based on core damage frequency (based on the latest NRC SPAR models) and initiating events involving containment bypass or leading to an early failure of the containment with potential consequences. Scenarios (sequence groups) with a frequency at or above 10^{-6} per reactor-year were screened in, as well as scenarios with a frequency at or above 10^{-7} per reactor year for scenarios that are known to have the potential for higher consequences (e.g., containment bypass scenarios such as steam generator tube rupture and interfacing system loss-of-coolant accident (ISLOCA) initiators). Please note that these criteria were not rigidly or inflexibly applied. In addition, some scenarios were included even though they did not explicitly meet our screening criteria. For example, candidate

SOARCA sequences were examined with radiological release timing in mind (both the timing of core damage and the timing of containment failure) and considered for inclusion since a major impact on both early and latent cancer fatality risks is derived from the timing of the offsite release. The Peach Bottom short-term SBO scenario was included, even though it did not satisfy our selection criteria, because it has a more prompt radiological release and a slightly larger release.

The selected sequence groups for SOARCA were shown to be important in recent and past probabilistic risk assessments, particularly for the well-studied pilot plants of Peach Bottom Atomic Power Station (Peach Bottom) and Surry Power Station (Surry). The scenarios selected for SOARCA were compared to NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," to corroborate that important scenarios were captured. The selected scenarios are also representative of broad classes of transients. The selection of station blackout (SBO) events in SOARCA ensures that we have covered that broader class of transients involving a loss-of-heat removal and further, by including a short-term station blackout (STSBO) we have reasonably bounded that class of accidents (which could include other events such as loss of service water or loss of component cooling water but which develop more slowly). Also, for the PWR, the station blackout also includes, in part, the effect of a small loss of coolant by considering reactor coolant pump seal leakage. In addition, by the selection of station blackout sequences for analysis, we also include the effects of loss-of-containment heat removal (fan coolers) and loss-of-containment spray systems (which are all electrically powered) to remove airborne radionuclides.

The purpose of SOARCA was to do a detailed best estimate consequence analysis. Pursuing a Level 3 PRA which would capture consequences from additional lower order sequences would necessitate introducing conservative assumptions in order to complete analyses and would have served a different purpose. The SOARCA study is not a traditional risk study which is intended to quantify total risk from all sequences, but instead focuses on quantifying the consequences of important severe accident scenarios.

In addition, the most recent Level-3 PRA-type information that we have, while somewhat simplistic and approximate, indicates that SOARCA is likely to have considered accident scenarios that account for the majority of risk at both the Peach Bottom and Surry plants. The Severe Accident Mitigation Alternatives (SAMA) analysis submitted with the license renewal application (2001) for Peach Bottom indicated that station blackout scenarios accounted for the majority of offsite risk (Reference: Peach Bottom Atomic Power Station, License Renewal Application, Environmental Report, Appendix G, available at: http://www.nrc.gov/reactors/operating/licensing/renewal/applications/peach-bottom/peach_bottom-envg.pdf). The SAMA analysis for the Surry license renewal application (2001) indicated that SGTR and ISLOCA accounted for the majority of the off-site population dose risk (Reference: chapter 5 of NUREG-1437, Supplement 6 (2002), "Generic Environmental Impact Statement for License Renewal of Nuclear Plants: Regarding Surry Power Station, Units 1 and 2 – Final Report," available at: http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1437/supplement6/2671bpart2.pdf).

2. **Page 10**

Comment:
The SOARCA analysis and report is developed by applying a method to two specific plants Surry and Peach Bottom. The application of the methods to two specific plants has both positive and negative aspects. The positive aspects are that with plant specific information, plant specific conclusions can be drawn and can be based on the specific design features, maintenance and operation practices at that particular site. The downside to this approach is that not all the plant specific features, both those features that reduce consequences as well as those that might increase consequences, are represented in the two plants chosen. As such, some conclusions are likely applicable to that site only and the results may not be typical.

Resolution:
The SOARCA analyses are indeed plant specific evaluations, and should not be used out of context. The team is aware that caution should be taken in projecting these results to other similar plants; however, insights from these analyses are generally expected to be useful for other plants. The appropriateness of applying individual insights to other plants should be evaluated on a case-by-case basis. This point will be included in the conclusion section of the main report.

3. **Page C-2**

Comment:

The objectives of the SOARCA project appear in several locations. In some of these locations the wording is slightly different. It is recommended that a single list of goals and objectives be developed and used consistently.

Resolution:

The document is being revised to refer to objectives in a consistent manner. Preliminarily, the objectives in the current draft are:

The overall objective of the State-of-the-Art Reactor Consequence Analysis (SOARCA) project is to develop a body of knowledge regarding the realistic outcomes of severe reactor accidents. Corresponding and supporting objectives are as follows:

- Incorporate the significant plant improvements and updates not reflected in earlier assessments including system improvements, training and emergency procedures, offsite emergency response, and recent security-related enhancements described in Title 10, Section 50.54(hh) of the *Code of Federal Regulations* (10 CFR 50.54(hh)) as well as plant updates in the form of power uprates and higher core burnup.

- Incorporate state-of-the-art integrated modeling of severe accident behavior that includes the insights of some 25 years of research into severe accident phenomenology and radiation health effects.

- Evaluate the potential benefits of recent security-related mitigation improvements in preventing core damage and reducing an offsite release should one occur.

- Enable NRC to communicate severe-accident-related aspects of nuclear safety to stakeholders including Federal, State, and local authorities; licensees; and the general public.

- Update quantification of offsite consequences found in earlier NRC publications such as NUREG/CR-2239, “Technical Guidance for Siting Criteria Development.”

4. Page C-2

Comment:

The abstracts in the reports are not used as effectively as they could be. Formal abstracts will be the location where the authors can summarize their findings, results and conclusions, and methods. These are important aspects of the report and it is recommended that they be fully developed.

Resolution:

The abstracts in the report will be reevaluated with the reviewer's suggestions in mind, and revised as necessary to ensure a high-level summary of findings, results and conclusions is captured.

5. **Page C-2**

Comment:

In addition, Executive Summaries are also not well utilized. Additional care could make them more effective.

Resolution:

The SOARCA team is in the process of revising the Executive Summary.

6. **Page C-2**

Comment:

Seismic research issues and the treatment of seismic issues have the general impression that their contribution would be a foregone conclusion. The area of seismic sequence conclusions reached are valid but the uncertainties associated with the occurrence of large seismic events as well as the consequences of such an event are high. This should be acknowledged in the report. In addition, on-going seismic research efforts should also be addressed or acknowledged.

Resolution:

Additional text was added to Appendix B Section 3.5, "Surry Seismic PRA (SPRA) Study," comparing the recent Surry Seismic Probabilistic Risk Assessment Pilot Study with the events that occur in SOARCA. The Surry SPRA study produced a total CDF which is comparable to the CDF of the SOARCA external events (on the order of 2 x 10-5/yr). The dominant scenario (comprising 50% of the total CDF) was identified as a loss of service water (LOSW). While consideration of such an event was not considered in SOARCA, the long-term SBO (LTSBO) scenario may qualitatively serve as a surrogate for the LOSW event in certain respects. In both instances, the plant is undergoing a loss of heat removal transient with potential reactor coolant pump seal leakage. In addition, the Surry SPRA study assumes the emergency condensate storage tank (ECST) and the fire protection water tanks are not available due to their low capability to withstand seismic loading. If indeed these tanks are assumed to fail catastrophically this would result in immediate loss of auxiliary feedwater (AFW) unless other sources of water for the AFW system can be aligned. It is clear that the worst-case LOSW scenario as identified in the Surry SPRA study combined with the loss of the ECST would still be bounded by the unmitigated short-term SBO scenario, since that scenario credits no primary side injection or auxiliary feedwater. The SOARCA team plans on discussing this comparison at our next Peer Review Committee meeting.

NRC is currently conducting ongoing seismic research, some of it in cooperation with EPRI under the NRC-EPRI Memorandum of Understanding. However, as this was not the focus of the SOARCA study, this ongoing research is not highlighted in the SOARCA reports.

7. **Page C-2**

Comment:
Several factors that anecdotally support the conclusions of lower consequences in SOARCA include changes to the physical plant and procedures. Some of these changes include the Station Blackout Rule, the ATWS Rule, development of Emergency Operation Procedures (EOPs) and Abnormal Operating Procedures (AOPs), plant specific simulators, severe accident management guides (SAMGs), the maintenance rule, and overall improved performance. These should be mentioned in the report.

Resolution:
The Station Blackout Rule, ATWS Rule, and Maintenance Rule are mentioned in the Main Report in Section 2.5, “Generic Factors,” as factors that contribute to the decrease of risk estimates for nuclear power plants over the years. EOPs are discussed in Section 2.2, “Scenarios Initiated by Internal Events,” of the Main Report and SAMGs and 10 CFR 50.54(hh) mitigation measures are discussed in Section 3.1, “Site-specific Mitigation Strategies,” of the Main Report. EOPs and SAMGs (as well as other 10 CFR 50.54(hh) improvements for

mitigated scenarios), and how they are incorporated into the analysis, are now discussed at the beginning of Section 2, “Accident Scenario Development,” in Appendices A & B.

CLÉMENT

April 30, 2010 SOARCA Peer Review Draft Report

8. **Page 13**

Comment:

Concerning the accident progression for Surry, one of the most important results of the analysis is that a creep rupture of the hot leg nozzle occurs before induced failures in other locations of the RCS and before failure of the lower head of the reactor pressure vessel. It is also considered that the rupture of the hot leg nozzle results in a large break. This has important consequences for what happens next. First, the depressurization of the RCS allows injection of water by the accumulators that delays the progression of the accident. Secondly, this avoids any high pressure melt ejection. In addition to this base case, scenarios with thermally-induced steam generator tube rupture were considered. Although the base case scenario is credible and corresponds to the best-estimate philosophy of SOARCA, uncertainties on different failure modes and locations must be taken into account.

Related Comment (p. C-3)**:**

The answer to comment #1 is not satisfactory, as no variability was introduced in the timing of RPV lower head failure. This could be done, as discussed at the last meeting, through a sensitivity study on parameters governing the relocation of corium to the lower head.

Resolution:

While hot leg rupture is predicted deterministically prior to thermally induced-steam generator tube rupture (TI-SGTR), an induced tube rupture sequence was selected as a variant of the Surry Station Blackout analysis. The most significant competing challenges occur between hot leg (HL) failure versus a TI-SGTR; the parameters that govern the timing of HL creep rupture relative to the TI-SGTR were examined. Section 5.3.3, “Uncertainties in the Failure of the Thermally-Induced Steam Generator Tube vs. Hot Leg,” in Appendix B examines the sensitivity of the timing of hot leg failure to the TI-SGTR. It was the conclusion of the current MELCOR analyses and the extensive number of analyses supporting NUREG/CR-6995, “SCDAP/RELAP5 Thermal-Hydraulic Evaluations of the Potential for Containment Bypass During Extended Station Blackout Severe Accident Sequences in a Westinghouse Four-Loop PWR,” that failure of these piping components (i.e., the HL and pressurizer surge line piping and SG tubes) is the most likely failure locations. The piping failures are encountered shortly after the time when the oxidation power peaks. Melting and relocation of the core fuel to the RV lower head are seen to occur long after the time of the piping failures.

Relative to the Dr. Clément's comment, the potential for other failure modes and locations must be taken into account. Based on considerable research including analysis of accidents, experiments, system code calculations, and CFD simulations, the most likely locations for RCS failure include: (1) the HL nozzle; (2) the steam generator tubes; (3) the pressurizer surge line; and (4) the lower head[1]. As discussed above, analyses of failures at the HL nozzle and the steam generator tubes were considered. Due to the presence of reactor coolant pump seal leakage, surge line failure was very unlikely because the seal leakage (and/or seal failures) prevented PORV cycling during core damage[2]. The final RCS failure location would be the lower head prior to other substantial RCS failures (i.e., if unmitigated, all scenarios eventually lead to lower head failure). For example, in the MELCOR sensitivity calculation where hot leg failure was prevented (see Appendix B Section 5.3.3), vessel failure was predicted to occur at 5.2 hours, nearly 2 hours after the TI-SGTR. During the 2 hours following the TI-SGTR, the hot leg creep rupture index exceeded the failure criteria by four orders of magnitude. A realistic combination of events or uncertainties has not been identified that would decrease or disable hot leg or whole loop natural circulation and the associated RCS structural failures (i.e., HL or TI-SGTR) for such a long duration. The basis for this conclusion also considered the results of NUREG/CR-6995, which includes the results of extensive code parameter variations, equipment response uncertainties, and operator actions. While conceptually possible, it was considered incredible for the scope of the present study (e.g., the same reason that an alpha mode failure was not considered).

9. Page 14

Comment:

The analysis shows that hydrogen combustion by jet ignition becomes possible after the hot leg rupture. Bounding cases are given for adiabatic, isochoric, complete combustion (AICC) and

[1] In addition to these locations, recent research by industry and NRC suggests that thermal failure of the instrument tubes leading to release into the containment is another potential location. Although this may have some significance in the B&W once-through steam generator design, it would occur much later than HL failure or TI-SGTR in a U-tube Westinghouse steam generator and therefore is not relevant.

[2] If the pressurizer surge line failed, the subsequent response is similar to a HL failure. Prior to the inclusion of RCP seal leakage and/or seal failure (i.e., circa 2000), the pressurizer safety valves continued to cycle during core damage, which preferentially induced hot gas flow from the core into the surge line. This led to failure of the surge line as dominant failure location.

detonation. It would be interesting to see if we are far or not from the σ criterion for flame acceleration and the λ criterion for detonation in order to evaluate.

Related Comment (p. C-3):
This comment refers to the presentation made by KC Wager at the last meetings. It is stated that a jet ignition is likely after hot leg creep rupture failure. Bounding cases are given for AICC and detonation. It would be interesting to see if we are far or not from the criterion for flame acceleration and the criterion for detonation. Those are given in the following document: "W. Breitung et al., "OECD State-of-the-Art Report on Flame Acceleration and Deflagration-to-Detonation Transition in Nuclear Safety," NEA/CSNI/R (2000)7, August 2000."

Resolution:
To address uncertainties in the STSBO for hydrogen combustion, the SOARCA team updated Section 5.2.3, "Uncertainty in the Hydrogen Combustion in the Mitigated STSBO," in Appendix B which reports the results of sensitivity analyses undertaken in response to peer reviewers' concerns. Based on the sensitivity analyses, the SOARCA team concludes that the best-estimate response reported in Section 5.2.2, "Mitigated Short-Term Station Blackout," is a reasonable representation of the source term, and any uncertainties in the best-estimate response should not have a material effect on the best-estimate results.

Section 5.2.3 describes the analysis of the potential for jet ignition following hot leg failure and vessel failure. In addition, a sensitivity study was performed that precluded any combustion until bulk detonation conditions were achieved (i.e., for the case with emergency spray mitigation). The adiabatic isochoric complete combustion (AICC) model for deflagration and the Chapman-Jouguet (CJ) model for detonation were used to assess the resultant pressure challenge (i.e., the results showed the peak pressure would exceed the best-estimate containment failure pressure). Additional 2-dimensional shock calculations were also performed. The reviewer recommends additional research to further evaluate the potential for flame acceleration from deflagration to detonation. At the time of hot leg failure, a hydrogen rich jet exits the hot leg. The bulk hydrogen values calculated at this time were relatively low and the steam content was high. Consequently, it was concluded that there was insufficient hydrogen to support the propagation of a detonation. Later, after emergency spray operation and further hydrogen production, there is high hydrogen and low steam content in the containment but the airborne fission products were knocked down by the same sprays that decreased the steam inerting potential. Future work may be appropriate to apply the referenced methodology to assess the likelihood of deflagration to detonation transition. But such future work is outside the scope of the current SOARCA study. As noted above, based on the sensitivity analyses, the SOARCA team concludes that the best-estimate response reported for Surry in Appendix B is a reasonable representation of the source term, and any uncertainties in the best-estimate response should not have a material effect on the best-estimate results.

10. Page 14

Comment:
No transport of gaseous iodine in the RCS is considered although this was experimentally evidenced. There is also no treatment of gas iodine chemistry in the containment. The Project made a sensitivity study to cope with this modeling lack: gaseous iodine concentrations observed in the Phebus FPT-1 experiment were added to the containment inventory. Phebus was an NRC funded project using MELCOR code conducted at Sandia National Labs. As the calculated iodine releases are already high, this addition does not make a big difference. It should however not be forgotten that this would probably not be true for other sequences with lower releases. Also, it is expected that gaseous iodine releases due to gas phase chemistry phenomena in the containment could last for a longer time that the 48 hours considered in the studies.

Resolution:
The SOARCA team acknowledges that gaseous iodine remains a source term issue, especially with respect to long term containment performance issues after the comparatively much larger airborne radioactivity has settled from the atmosphere. We consider the mechanistic modeling treatment for gaseous iodine behavior to be a technology still under development with important international research programs underway to determine the dynamic behavior of iodine chemistry with respect to paints, wetted surfaces, buffered and unbuffered water pools undergoing radiolysis and gas phase chemistry. We believe that the base case treatment under our best practices recommendation is sufficient for the best-estimate effects addressed in SOARCA, and we plan to investigate parameterization of the gaseous iodine fraction of total iodine releases in the uncertainty analysis for the project.

11. Page 15

Comment:

Progress has been made in the recent years in the knowledge of accident progression and source term evaluation. Not all the outcomes have been incorporated in MELCOR models and advances in knowledge are still ongoing. It should be valuable, when a MELCOR version incorporating significant new features becomes available, to benchmark the present SOARCA results with this new version for some selected sequences.

Resolution:

There would be value in reassessing such studies as new significant source term information becomes available. Nonetheless, the SOARCA team does feel the SOARCA analyses are consistent with the present state-of-the-art.

12. Page C-3

Comment:

Synthesis report pp. 11-12 – Some words could be added about the uncertainties on accident progression. Not only the weather conditions and their consequences will be considered in the uncertainty analyses.

Resolution:

Revised Section 1.9, “Uncertainty Analysis,” now includes additional discussion of uncertainties in accident progression. In addition, although uncertainties will be addressed in the uncertainty analysis in an integrated fashion, many sensitivity studies have been conducted since the March 2010 Peer Review Committee meeting that investigated uncertainties identified in the accident progression analysis. The results of many of these sensitivity analyses are included in the SOARCA reports, and some of these will be discussed at our next Peer Review Committee meeting.

13. Page C-3

Comment:

The answer to comment #4 by Clement clarifies the use of CDF as screening criteria. It would be valuable to add this text in the final report as well as parts of answer to comment #5 and to comment #58 by Leaver.

Resolution:

Chapter 2, "Accident Scenario Selection," of the Main Report is a detailed presentation of the accident scenario selection process, and emphasizes the key points in the answers referenced in this comment.

14. Page C-3

Comment:
The treatment of comment #3 gives a correct answer, showing low consequences on RN releases. It should however be good, in the future, to consider a distribution of initial defects in the SG tubes, obtained from inspections' feedback experience.

Resolution:
A considerable amount of work has been done by the NRC analyzing the potential for thermally induced steam generator tube rupture (TI-SGTR) (e.g., NUREG/CR-6995). SOARCA incorporated the findings from these studies to include the potential for TI-SGTR. Two cases were considered: a single tube rupture and two tubes. The failures were assumed to occur near the steam generator inlet plenum tube sheet where the high temperature stream enters the tube bundle. The one and two tube TI-SGTR cases showed interesting and divergent effects of enhancing oxidation and providing additional core cooling, respectively. As the reviewer notes, additional realism could be introduced by reviewing data from plant inspections to examine the location and magnitudes of defects. This data could be cross-correlated against the CFD work done by the NRC (i.e., NUREG-1788, "CFD Analysis of Full-Scale Steam Generator Inlet Plenum Mixing During a PWR Severe Accident") against the likelihood of those tubes receiving the highest temperature flow stream from the inlet plenum. In effect, it was assumed that the most vulnerable defect(s) was exposed to the highest temperature gas stream entering the SG. Since the high temperature stream cools rapidly as it flows through the steam generator, the timing of the failure for other locations would only decrease the timing between the subsequent hot leg failure. It was believed that the SOARCA approach was conservative in this respect but consistent with the current state-of-the-art. Additionally, separate multi-year work is underway, in a different NRC project, to assess the likelihood of TI-SGTR, considering updated flaw distributions and material changes. Future analyses can benefit from additional realism by considering inspection experience.

JEFF GABOR

April 30, 2010 SOARCA Peer Review Draft Report

15. Page 16

Comment:
Due to the primarily deterministic approach taken, great care must be taken in communicating these results in any context that include a discussion of risk to the public. The project and associated documentation details a more realistic assessment of the potential consequences associated with operating nuclear reactors for the accident progression scenarios evaluated and portrays a more up-to-date understanding of the key accident phenomena.

Resolution:
The SOARCA team has a separate explicit task from the Commission to ensure that SOARCA is communicated effectively to the public. The importance of accurately portraying the results of this study and its implications on risk is understood by the SOARCA team. This concern regarding communication results has been expressed by other peer reviewers too, and is addressed in other responses as well. The SOARCA study is not a traditional risk study which is intended to quantify total risk from all sequences, but instead focuses on quantifying the consequences of important severe accident scenarios. The study does, however, provide quantification of individual health risk for the accident scenarios selected.

16. Page 17

Comment:
As the SOARCA project did not evaluate a full spectrum of scenarios, great care must be taken in the communication of these results. While potentially representative, these results are plant-specific, limited in scope, and do not fully characterize plant risk.

Related Comment (December 3 Memo, p. 2)**:**
Care should be taken in characterizing SOARCA in the context of a “risk” study. Since final documents have not been provided to the Peer Review Team, it is not confirmed that this has been addressed.

Resolution:

This aspect of the study has been a subject of ongoing discussion. Unlike PRA's such as the NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," study or contemporary PRA's, whose objective is to address a fuller risk profile, SOARCA is intended to be an assessment that identifies and quantifies consequences (and scenario specific risk) from the more important accident scenarios. We believe that the detailed examination of the specific scenarios selected in SOARCA provide a realistic view of potential consequences. See also response to Mr. Canavan's Comments 1 and 2.

17. Page 17

Comment:

Lower Head Penetration Failure – Comments were provided as to the omission of lower head penetration failure as a possible vessel failure mode. The SOARCA analysis did not include these failure mechanisms based on the fact that the majority of BWR accident sequences are assumed to result in the RPV being depressurized prior to core relocation into the lower head. It is acknowledged that the likelihood of these failure mechanisms is reduced at lower RPV pressures.

Related Comment (p. C-4):

[MELCOR Best Modeling Practices] Section 3.1.1.5 - I would recommend a little more explanation on why penetration failure as a mode of vessel breach has been ignored. This needs additional justification.

Related Comment (December 3, 2010 Peer Review Memo, p. 2)**:**

BWR lower head penetration failures are not represented. Clear justification has not been provided.

Resolution:

BWR penetration failure is considered to be a complex failure mode. At the time of material relocation to the lower head, both the lower head and the guide tubes are filled with water. It is assumed that this water must first be evaporated before significant thermal attack occurs. If guide tube failure occurs prior to lower head failure, there is still no clear pathway for movement of core materials out of the vessel region, owing to the presence of drive tube mechanisms and considerable structural materials residing in the drive tube below the vessel head. Concerning the lower head drain plug, we feel that this penetration will likely be plugged by lower melting point control blade materials and will not be predominantly affected by subsequently relocating core materials. Owing to the complexity of modeling this detail, we elected to model only the vessel head structure itself. Without experimental information to guide such complex modeling, the approach taken in the SOARCA project is considered commensurate with the current state of knowledge. As noted in Section 4.3.1, “Peach Bottom MELCOR Model,” of the Main Report, a penetration failure model was not used because the timing differences between gross lower head failure and penetration failure with the available penetration model are not significant to the overall accident progression (i.e., minutes difference). Also, Sandia lower head failure tests showed gross creep rupture of the lower head was measured to be the most likely mechanism for vessel failure (Chu, T.Y., et al., NUREG/CR-5582, SAND98-2047: Lower Head Failure Experiments and Analyses. 1998, Sandia National Laboratories: Albuquerque, NM.).

18. Page 18

Comment:

Hydrogen ignition in SBO - comments were provided to identify the source for hydrogen ignition in the station blackout sequences. Section 5.1.3 of the Surry Accident Analysis was updated to include a more thorough discussion of ignition sources. Hot gases exiting the reactor vessel upon hot leg creep rupture and at the time of lower head failure were shown to have sufficient energy to ignite the hydrogen. An additional investigation was performed to study hydrogen combustion upon mitigation using containment sprays. Prior to spray recovery the containment atmosphere can be inerted by the steam present; however, as the steam fraction is reduced from spray actuation, small burns are shown to occur. My review comment addressed a possible delay in hydrogen ignition upon spray actuation and Section 5.1.3 was revised to include this sensitivity.

Resolution:

It is the SOARCA team’s interpretation of this comment that the reviewer is satisfied with the revised additional analysis of the spray actuation and hydrogen ignition.

19. Page 18

Comment:

SRV failing in the open position- the SOARCA analysis identified SRV sticking open during core heat-up as the dominant mechanism for causing RPV depressurization. Competing phenomena includes the heat-up and potential failure of the Main Steam Line nozzle. As a result of my comments, Section 5.6 of the Peach Bottom Integrated Analysis includes a substantial analysis of the uncertainty associated with the SRV failure mode. Cases were included assuming an early failure of the SRV, a failure but with only ½ of the relief area, and a case without SRV failure but with subsequent creep failure of the main steam line nozzle. These sensitivity cases provide valuable insights and show that the highest release of iodine to the environment is associated with the MSL creep failure case. Where it is understood that the SOARCA development team believes that SRV failure case represents the best-estimate, it would be useful to show the consequence impact due to the MSL failure case. In addition, the impact of the hot gas on the potential for Drywell head failure resulting from the MSL failure was not considered. The sensitivity of the results to this failure mode are further evidence that focus on the analysis and reporting of individual accident progression scenarios can be misleading. This is why a fully risk-informed approach to the presentation of consequence information is preferable.

Related Comment (December 3 Memo, p.2)**:**
Potential for Main Steam Line failure not justified. Where there has been a sensitivity case provided for MSL creep failure, the impact on consequences has not been calculated. There needs to be a strong technical case developed for why there will not be a MSL creep failure. Even with a lower expected probability for MSL creep failure, higher consequences could make this important.

Resolution:
In order to address Dr. Gabor’s comment on how the SOARCA team believes the SRV failure case is a best-estimate, the SOARCA team reanalyzed the SRV cases identifying two conditions that would lead to the MSL creep rupture. Results of these MELCOR calculations are described in Section 5.5.2, “Extreme Variations in Failure Criteria,” of Appendix A. These MELCOR calculations determined that the following conditions are necessary for MSL creep rupture: 1) stochastic SRV failure being ignored, and over temperature failure of an SRV not being considered until valve stem temperature reached a very high value (1,175K); or 2) thermal failure occurs at the best-estimate temperature criterion, but the resulting open fraction of the SRV is very small (10% or less). These conditions are judged to be extremely unlikely given the current understanding of SRV failure, and thus were not considered representative of the best-estimate analysis. Therefore, no consequence calculations were conducted. Many sensitivity

studies were performed exploring a wide range of means by which MSL failure would occur. Please note that this issue will also be looked at as part of the uncertainty analysis.

20. Page C-4

Comment:

I believe Dr. Henry previously identified this, but it would be good to include a discussion of the differences between BWR and PWR core. This could be added to [MELCOR Best Modeling Practices] Section 3.1.3 and simply explain the differences (channel boxes, etc.) and provide some discussion of their impact.

Resolution:

Section 4.1, “Reactor Vessel and Coolant System,” and 4.2, “In-vessel Structures and Reactor Core,” of Appendix A provide information on the BWR reactor and core configuration and Section 4.1, “Vessel and Reactor Coolant System,” of Appendix B provides information on the PWR. From these two appendices a reader can identify the differences between a BWR and a PWR core. More information will be provided in these sections to discuss how unique features of both types of core designs are modeled in MELCOR.

21. Page C-4

Comment:

[MELCOR Best Modeling Practices] Section 3.1.1.5 – I would also recommend some discussion of structures in the lower plenum (instrument tubes, CRD tubes, etc.) and an indication of what their impact would be. This is another area where differences between BWR and PWR could be highlighted.

Resolution:

As noted in the response to Comment 20, additional information will be provided in Sections 4.1, “Reactor Vessel and Coolant System,” and 4.2, “In-vessel Structures and Reactor Core,” of Appendix A to discuss the manner in which unique design features of the BWR (versus PWR) lower plenum are modeled in MELCOR.

22. Page C-4

Comment:

[MELCOR Best Modeling Practices] Section 3.1.1.6 – I would recommend some discussion of the impact of structures in the cavity area on debris spreading and cooling.

Resolution:

The SOARCA team is not aware of any structures within the BWR reactor pedestal that would significantly impede the flow of debris, except for the pedestal wall itself. An open doorway from the pedestal to the main drywell floor restricts (or directs) flow from the pedestal to the drywell floor to a small fraction of the azimuthal circumference of the floor. But this restriction is accounted for, as described in Section 4.5, “Behavior of Ex-Vessel Drywell Floor Debris,” of Appendix A.

23. Page C-4

Comment:

[MELCOR Best Modeling Practices] Section 3.1.1.7 – I recommend an explanation of why they assume a PWR valve will fail at the cumulative failure probability of 50% and a BWR valve at 90%.

Resolution:

The PWR analysis selected 50% failure probability from the beginning of the SOARCA analysis to represent median failure conditions. A 90% failure probability was used in the initial BWR calculations to represent a "high confidence" level for an event that was perceived to be a ‘benevolent failure,’ That is, a failure that would lead to a more delayed and smaller source term. These different modeling approaches developed independent of each other, and the inconsistency was recognized later as a consequence of questions raised by Peer Review Committee. When the SOARCA analysis was revised to address these (and other) peer review comments, the differences in failure criteria narrowed, but were also found to be unimportant to the results, as explained below.

The approach used to model stochastic failure of an SRV to reclose in the BWR analysis was replaced by a more 'best estimate' approach based on early peer review comments. The revised criterion for stochastic SRV failure was defined based on the "expected value" for the number of cycles a valve would experience at the time of failure. ‘Expected value’ is calculated as 1/failure-rate. If one translates this approach to a cumulative probability at the time of failure, the value corresponds to a 63% confidence level for BWRs, which is closer to, but still different from, the (assumed) 50% probability used in the PWR analysis. The calculated number of cycles experienced by primary and secondary coolant system relief and safety valves in the PWR is much less than the number corresponding to the median (50%) failure probability. Therefore, stochastic failure never occurs in the PWR calculations. Confidence in this observation would only increase if the failure condition were shifted from the median failure probability to the probability corresponding to the "expected value" (63%). Therefore, an adjustment to the PWR model was not deemed necessary. A footnote will be added to Appendix B to explain that the exact cumulative failure probability chosen did not matter for Surry.

24. Page C-4

Comment:
[MELCOR Best Modeling Practices] Section 3.1.3.1 – This section needs to discuss Drywell shell failure. Section 4.3 even points here for such a discussion.

Resolution:
Additional discussion of the relationship between lateral debris mobility and drywell shell melt-through as a containment failure mechanism have been added in Appendix A Sections 4.5, "Behavior of Ex-vessel Drywell Floor Debris," and 4.6, "Containment Failure Model."

25. Page C-4

Comment:
[MELCOR Best Modeling Practices] Section 4.2 – For completeness, DCH in a BWR should be discussed and reasons for it being a low threat included.

Resolution:
MELCOR Best Practices Section 4.2, "Direct Containment Heating (DCH)," includes the discussion of DCH in PWRs. DCH in BWRs has not been as extensively investigated as with PWRs. However, similar to PWRs, opportunities for depressurization in the reactor vessel prior to that DCH event exist, such as operation of the safety relief valve system (SRV), or sticking or seizure of the SRVs. Focused studies within the SOARCA project conclude that SRV seizure open is very likely to occur well before any potential direct containment heating event. A short discussion on DCH in BWRs will be added to the MELCOR Best Practices document.

26. Page C-4

Comment:

The end [Appendix A] Section 4.5 raises “drywell liner melt-through” as one of the containment failure modes considered. It points the reader to Section 4.4, however, there is no discussion on liner melt-through. I recommend that there is a brief statement on what liner melt-through is and what the assumed criteria for failure is. It is clearly stated that water will prevent it, but no details are ever provided on what the failure model/criteria is. It might also be helpful to indicate the assumed area of failure and maybe a discussion of the release of pathway associated with failure mechanism.

Resolution:

New text has been added to Sections 4.5, “Behavior of Ex-Drywell Floor Debris,” and 4.6, “Containment Failure Model,” of Appendix A, as noted in the response to Comment 24.

27. Page C-4

Comment:

[Appendix A] Section 5.2 – LTSBO discussion: I recommend a statement on the assumed operator action to vent the containment. It only shows up on the figure with no discussion. PCPL is closer to 60 psia, so venting at 40 psia needs to be explained and perhaps a description of the “possible” release pathway. I just think that this action needs to be called out in the text somewhere.

Resolution:

The principal basis for the containment venting criteria used in the original calculation of the mitigated long-term station blackout (LTSBO) (reviewed by the Peer Review Committee) was comments received from the licensee during a verbal walk-through of LTSBO mitigation, which was part of a site visit in 2007. Based on this comment (and others) received from the Peer Review Committee, the criteria were re-examined and a new (replacement) calculation has been performed, which is identical to the earlier one except for the assumed criteria for opening/closing the hard-pipe containment vent line. The new calculation assumes the hard-pipe vent path was opened at 45 psig and reclosed at 25 psig. These values were selected based on a review of plant-specific procedures for containment pressure control (Peach Bottom procedure T-102), but also taking into consideration isolation setpoints for RCIC, which are sensitive to containment thermodynamic conditions. Most important in this regard is the setpoint for high turbine exhaust pressure. The Peak Containment Pressure Limit (PCPL) suggested in procedure T-102 is 60 psig. However, a high turbine exhaust pressure isolation signal for RCIC would be received at a pressure of 50 psig. Since RCIC is the only operating coolant injection system available in this scenario, we assumed operators would open the containment vent path at 45 psig, thereby averting RCIC isolation.

Results of the revised calculation are the same as those obtained in the earlier calculation except for the containment pressure response. Discussion has been incorporated into the updated version of Appendix A.

December 3, 2010 Peer Review Memorandum

28. Page 2

Comment:
Most aspects of SOARCA represent a plant-specific evaluation and should not be extrapolated to other plants.

Resolution:
SOARCA analyses are indeed plant specific evaluations, and should not be used out of context. The team is aware that caution must be taken in projecting these results to other similar plants; however, insights from these analyses may be useful for other plants. The appropriateness of individual insights for other plants must be evaluated on a case-by-case basis.

29. Page 2

Comment:
For mitigated Surry SBO, the initiation of containment sprays may result in an early containment failure due to hydrogen combustion. A clear justification has not been provided for why this will not be a significant contributor to off-site consequences.

Related Comments (received before April 2010):
Mitigated short term SBO: why are there H2 burns? Is there a criterion for ignition when there is no power? Is nodalization controlling? What would be the impact of delaying the burns due to inadequate ignition?

Hydrogen burn (deflagration) was discussed, but there was no discussion of hydrogen detonation. Has this been evaluated to be below the CDF defined? In this reviewer's experience, hydrogen detonation, depending on their size and location, can cause large leakage or breach of containment

Resolution:
A number of hydrogen combustion and detonation studies were conducted as part of the SOARCA analysis. Only delayed hydrogen combustion was shown to be a threat to containment integrity. In this case the radionuclide concentration in the containment environment was too low to result in a significant environmental release.

Section 5.2.3, "Uncertainties in the Hydrogen Combustion in the Mitigated Short-Term Station Blackout," of Appendix B was updated and examines uncertainties in the time of combustion and the impact of hydrogen detonation. No additional consequence calculation was performed on these calculations because the source term was limited to noble gases.

See the resolution to Comment 9 for further discussion.

30. Page 2

Comment:
Uncertainty analysis needs to be performed.

Resolution:
An uncertainty analysis is underway and draft results are expected to be available by mid 2012. We will be discussing the uncertainty analysis in more detail in a future Peer Review Committee meeting.

ROBERT HENRY

April 30, 2010 SOARCA Peer Review Draft Report

31. Page 21

Comment:

Throughout the report, there are numerous places where the "Objective" of the SOARCA assessment is defined. These all relate to the best estimate nature of the evaluations but the statements are not identical. For something as important as the objective of the study, the wording should be agreed upon and either be repeated exactly, or referenced, (to another part of the study), every place where this needs to be discussed. From my perspective, the important aspects of SOARCA are as follows:

- The central estimate/calculation of every aspect of the study is focused on the best estimate which is an appropriate focus for a state-of-the-art examination.
- This study is supported and directed by the Nuclear Regulatory Commission so it should be clearly stated that this study is specific to the U.S. fleet of commercial nuclear power plants. Clearly these are representative of a BWR and a PWR, with each having one of the important containment types used in the U.S.
- The studies include several plant specific features associated with the RCS and containment design, EOPs, SAMGs, etc. Hence, this shows the important influence of several plant specific features that have been included as operator actions, etc. that are taken during the accident progression.

Therefore, I suggest that the objective statement for the SOARCA be something like what is in the Abstract of the Summary document, but with some additional text. My suggestion is as follows:

The primary objective of the SOARCA project is to provide a best estimate evaluation of the likely consequences of important severe accident events at reactor sites in the U.S. civilian nuclear power reactor fleet. To accomplish this objective the SOARCA project has applied integrated modeling of accident progression and off site consequences using both state-of-the-art computational analysis tools to two previously analyzed reactor sites (Peach Bottom and Surry). To meet the state-of-the-art objective, the analysis tools utilized best modeling practices drawn from the collective wisdom of the severe accident analysis community. Equally important, the analyses for both of the reactor sites also represented the implemented procedures in the main control room and elsewhere, that are relevant to the response for the *important accident conditions related to highly unlikely, but possible radiological releases.*

Resolution:
The document is being revised to refer to objectives in a consistent manner. Preliminarily, the objectives in the current draft are:

The overall objective of the State-of-the-Art Reactor Consequence Analysis (SOARCA) project is to develop a body of knowledge regarding the realistic outcomes of severe reactor accidents. Corresponding and supporting objectives are as follows:

- Incorporate the significant plant improvements and updates not reflected in earlier assessments including system improvements, training and emergency procedures, offsite emergency response, and recent security-related enhancements described in Title 10, Section 50.54(hh) of the *Code of Federal Regulations* (10 CFR 50.54(hh)) as well as plant updates in the form of power uprates and higher core burnup.

- Incorporate state-of-the-art integrated modeling of severe accident behavior that includes the insights of some 25 years of research into severe accident phenomenology and radiation health effects.

- Evaluate the potential benefits of recent security-related mitigation improvements in preventing core damage and reducing an offsite release should one occur.

- Enable NRC to communicate severe-accident-related aspects of nuclear safety to stakeholders including Federal, State, and local authorities; licensees; and the general public.

- Update quantification of offsite consequences found in earlier NRC publications such as NUREG/CR-2239, "Technical Guidance for Siting Criteria Development."

32. Page 22

Comment:
The inclusion of a MELCOR "best practices" document is a very important feature of the SOARCA evaluation. It defines the manner in which the accident progression for both BWRs

and PWRs was evaluated as part of these central estimate calculations and also provides some of the features that are to be explored through the upcoming uncertainty analyses. In that regard, it is necessary that the best practices document describes the manner in which the evaluations were performed. It is important that the review committee reviews and comments on the controlled features associated with the MELCOR calculations.

Resolution:
The SOARCA team agrees that the MELCOR 'best practices' is a very important feature of the present study. The team will present information to the peer reviewers at a planned forthcoming meeting and information will be included in the Peach Bottom and Surry Reports. Please note that the MELCOR Best Practices document is expected to be available by late 2012.

33. Page 22

Comment:
In the current draft, there is a good description on the manner in which "breakout" of molten zirconium through a thin layer of oxidized cladding is evaluated in the MELCOR code for these analyses. This relocation of metallic zirc is an important feature associated with the overall melt progression. In addition, there is an extensive discussion of the dominant chemical states of the fission products and how these are evaluated in terms of the release rates from the oxide fuel and into the high temperature gas space of the Reactor Coolant System (RCS). There is also an extensive discussion on the modeling approach for cesium molybdate release rates for the fuel. In the current version, much of this appears to be written as part of the PWR description. However, these features are common to all of the BWRs and PWRs in the U.S. commercial fleet and should be clearly described as such in the write-up.

Resolution:
Additional text was added to Appendix A to include the relevant discussion for BWRs. In addition, other features have been included to document the manner in which the central estimates were evaluated. See response to Comment 24 for a specific example. The MELCOR Best Practices report will also be revised accordingly and is expected to be available by late 2012.

34. Page 22

Comment:

The release fractions of the dominant chemical states provides the manner in which the fission products from the fuel become airborne in the core region. The transport of these fission products from the core, thorough the RCS and into containment, as well as their deposition in these regions is determined by the aerosol model. Typically, the aerosol densities within the reactor coolant system can be in excess of 100 grams per cubic meter, which is a very dense aerosol. Hence, the deposition within the RCS can be quite large and the manner in which this is calculated needs to be documented as part of the "best practices". I suggest that the benchmarks of the aerosol model with experiments such as the large scale ABCOVE tests, the DENONA test, etc., where available, be included in this "best practices" document. This is important to capture since the aerosol transport and deposition model is that feature of the calculation that determines the extent of airborne fission products in the containment that could be released to the environment. It is particularly key that this discussion be included, along with the benchmarks that are relevant to the aerosol densities typically encountered in the RCS and containment, to be assured that indeed a central estimate is justified.

Resolution:

Currently, it is planned that analyses demonstrating code model assessment and validation will be documented and maintained in the MELCOR Code Manual Volume 3. The last published version of this report was for MELCOR version 1.8.5 and it is being updated for code Version 2.1. Essential validation exercises for the most part are not strongly dependent on the code version, as MELCOR models are relatively mature at this point. For example, validation analyses were last performed on the ABCOVE and Marvekin tests with code Version 1.8.2 but will be revisited with MELCOR 2.1 in the near future. The importance of validating the aerosol mechanics, transport and deposition modeling is well appreciated, and considerable validation in this area presently exists for MELCOR, including the VANAM experiments (similar to DEMONA and also performed in the Battelle Model Containment facility), LACE, CSE, and smaller scale facilities such as FALCON and AHMED, where aerosol agglomeration, hygroscopic effects and deposition, settling and spray scrubbing models are assessed and validated. The following table summarizes the status of code validation tests, including aerosol tests, for various versions of MELCOR and the plans for future documentation. Some discussion concerning code version and validation status follows.

Assessment\Code Version	1.8.1 (1991)	1.8.2 (1992)	1.8.3 (1994)	1.8.4 (1995)	1.8.5 (1996)	1.8.6 (2005)	2.1 (2011-2012)	MELCOR 1.8.5 Volume 3	CONTAIN Parity	Phebus Synthesis	IBRAE Assessments	Separate Assessments[3]	MELCOR 2.1 Volume 3
FLECHT-SEASET	x						x				x	x	x
LACE-LA4							x	x	x				x
CSE Spray Experiments			x		x				x			x	x
Marvekin	x										x	x	
DEMONA		x										x	x
ABCOVE Tests		x					x					x	x
NUPEC Mixing Tests					x		x	x	x				x
Ahmed Hydroscopic Tests					x	x	x		x				x
VANAM-M3 (ISP37)					x		x	x	x				x
IET DCH Experiments					x		x		x				
HDR-V44					x		x		x				x
HDR E-11							x					x	x
FALCON Tests							x				x		x
NTS Hydrogen Burn Tests					x		x		x				x
BWR Mk-III Vent Clearing Tests				x	x		x		x				x
GE Level Swell Tests			x				x		x			x	x
PNL Ice Condenser test	x				x				x			x	
RTF Iodine Tests (ISP41)					x				x				x
BESTHSY (ISP-38)							x				x		x
PBF-SFD1-4							x				x		x
LOFT-FP2	x					x	x				x	x	x
TMI-2					x	x	x	x				x	x
Phebus B9+					x			x					x
Phebus FPT-1					x	x	x	x		x		x	x
CORA 13	x				x			x					x
ACRR MP-1/MP-2		x										x	
ACRR DF-4		x										x	
RASPLAV Salt Tests							x				x		
RAS MEI Tests							x				x		x
NEPTUNE Experiment							x				x		x
SURC MCCI							x				x		x
HI/VI FP Tests					x		x			x			x
VERCORS 1-6 & HT/RT FP Tests					x		x			x			x

[3] "Separate Assessments" means that there is a stand-alone report (SAND or NUREG report) that documents the work.

Aerosol mechanics for nonhygroscopic aerosols is modeled using the MAEROS code (analogous to the NAUA code) where good verification of aerosol agglomeration physics and gravitational depletion was demonstrated in early versions of MELCOR based on MARVEKIN, ABCOVE and LACE testing. MELCOR Version 1.8.5 introduced extensions to treat hygroscopic aerosol effects where good validation against the VANAM M3 test (similar to DEMONA) as well as the AHMED experiments was demonstrated. The CSE A9 test was used to validate the containment spray scrubbing modeling in MELCOR on code Version 1.8.5 in the CONTAIN-MELCOR parity assessment study. The CONTAIN-MELCOR parity study introduced numerous other containment behavior assessments including the NUPEC mixing tests, the Nevada Test Site hydrogen burn tests, and the IET DCH containment heating experiments. Fission product release from fuel, including MOX and High Burnup were assessed against ORNL HI/VI tests and against more recent VERCORS experiments and documented in the Phebus Synthesis report using MELCOR Version 1.8.5. In Version 1.8.5 fission product release models were adjusted using sensitivity coefficient over-rides to the Version 1.8.5 models. These were formalized as code options and defaults for code Version 1.8.6. MELCOR Version 1.8.6 also introduced expanded modeling detail for core melt progression processes, including molten pool convection treatments. These extensions provided improved prediction of the TMI-2 accident, some of which are still currently under assessment. The Phebus FPT-1 test stands as the most comprehensive integral assessment of core damage progression, hydrogen generation, fission product release and RCS deposition and containment natural depletion processes. This test provides good assessment of key deposition behavior in the reactor RCS and for containment depletion. MELCOR Version 2.1 is largely identical to Version 1.8.6 with respect to model pedigree, the main difference being conversion of the source code to FORTRAN 95. Other code assessments for code Version 2.1 have been performed by IBRAE during the code Version 2.1 conversion process as indicated in the table. The MELCOR 2.1 code manual Volume 3 will largely compile the amassed collection of code assessment problems. The MELCOR Best Practices report is intended to document the specific manner that the latest code version is being applied to specific applications such as SOARCA plant models. It is our strong contention that code Version 1.8.6 largely embodies the collective state of the art with respect to MELCOR severe accident progression models with specific plant application treatments documented in the current MELCOR Best Practices report (the SOARCA revision of the Best Practices document is expected to be available in late 2012).

The validation of code modeling, including aerosol deposition, will be discussed in more detail at the next Peer Review Committee meeting and documentation of that information will be incorporated into the SOARCA revision of the MELCOR Best Practices report.

35. Page 22

Comment:

The accident progression within the core region from the intact fuel rods with "breakout" of molten zirconium that drains to the lower core region, eventual relocation of the molten debris from the core to the lower plenum and the controlling heat transfer to the RPV lower head need to be described. With the differences in geometry between the PWR and the BWR, as well as some potential differences between the U.S. commercial fleet PWR designs, for example upflow versus downflow core bypass, this should be described in the "best practices" document since this will be consulted, evaluated and referenced in future studies. Furthermore, only through an understanding of this core melt progression and relocation to the lower plenum can the features that should be investigated through uncertainty analyses be clearly defined.

The general public is well aware of the severe core damage accident that occurred in the Three Mile Island Unit 2 reactor. Any integral thermal-hydraulic model that is used to assess the timing for the onset of core damage, the release of fission products from the core, the extent of hydrogen generated in the core degradation, the transport of molten core debris to the lower plenum, etc. needs to be benchmarked with this accident. This benchmark evaluation needs to be either part of the SOARCA documentation or, at the minimum, referenced extensively in the other SOARCA reports.

Resolution:

Much of this has been documented in the MELCOR Code Manual Volume 3 document and will be referenced in the current reports to allow the reader further access to the information. Central values of key MELCOR parameters are derived from the assessments in this document (MELCOR Volume 3). The MELCOR Best Practices report includes an overview of the numerous benchmark activities undertaken under the MELCOR project, and makes reference to the MELCOR code manual set, which includes Volume 3 (SAND2001-0929P) with MELCOR Version 1.8.5 which is a collection of benchmark and validations analyses that include TMI-2. The molten Zr "breakout" parameter, as well as other MELCOR COR-Package parameters affecting core melt progression such as rod-collapse criteria and effective UO_2/ZrO_2 liquefaction behavior is discussed in the MELCOR Best Practices. These parameters affect the amount of hydrogen generated, peak core-region temperatures and core melt progression timing and behavior, including melt relocation rate and freezing behavior. The central values for these important parameters are derived principally from comparison historically to CORA, PBF, ACRR DF-4 and LOFT FP-4 experiments, and more recently to the Phebus experiments, principally FPT-1 and the TMI-2 benchmark analysis. Assessment/validation against CORA, LOFT-FP2, Phebus FPT-1 (soon to add FPT-3), and TMI-2 are included in the MELCOR Volume 3.

36. Page 23

Comment:
Evaluating accident progression of severe accidents in BWRs and PWRs involves the physical modeling of many complicated and interrelated processes. Given that these are both complicated and interrelated means that there are numerous uncertainties that need to be considered in developing best estimate analyses. These uncertainties need to be identified in the documentation and their influence on the conclusions of the study must be included in the final assessment.

Resolution:
Sensitivity studies have been conducted since the March 2010 and October 2010 Peer Review Committee meetings that investigate uncertainties identified in the best-estimate analysis. The results of these sensitivity analyses have been included in the best-estimate reports, and these will be discussed at our next Peer Review Committee meeting. An uncertainty analysis is underway and we will be discussing this in more detail in a future Peer Review Committee meeting.

December 3, 2010 Peer Review Memorandum

37. Page 3

Comment:
The SOARCA best estimate analyses, and the ongoing uncertainty analyses, are needed by the NRC to remain current with the developing state-of-the-art for severe accident research and to effectively regulate the new designs that are to be submitted for certification. To ensure that the MELCOR accident sequence calculations represent best estimate evaluations, it is important to have the various models in the code benchmarked to experimentally observe results.

Resolution:

Please see the resolution to Comment 38.

38. Page 3

Comment:

Provide entire code benchmarks

- TMI-2
- Phebus

Individual models

- NRC/EPRI/Westinghouse Steam Generator Natural Circulation Tests
- MPA Stuttgart Hot Leg Creep Test
- SNL Lower Head Creep
- HDR E11.2 and T31.5 Tests
- ABCOVE Large Scale Aerosol Tests
- DEMONA Large Scale Aerosol Tests

Those denoted in the above list were specifically identified in my previous comments. It is important that the benchmarks utilize the same logic/methodology in establishing the nodalization that is used in the SOARCA plant calculations. Furthermore, the parameters varied in the uncertainty study should also be used in the benchmarks and the numerical values should be close to the central estimates for the "uncertainty variations."

The committee has asked for this background material in various ways and has yet to receive any such information. Computer code benchmarks that qualify the code calculations are the necessary foundation to the study.

Resolution:

It is currently planned that code benchmarks such as the ones cited by Dr. Henry, will be added to and maintained in the MELCOR code manual Volume 3 on Assessments and Benchmarks as established validation benchmarks for the MELCOR code. A summary of code benchmarks performed to date is provided in a table in the response to Comment 34. In comparison, the "Best Practices" document is intended to be a living document (with specific dated versions) that describes what is deemed to be our current best modeling practices regarding model inputs that may be different from the default values in the last published MELCOR User Guide Manual, or that reflect user inputs that are unique to the system being modeled. An example is the user input reflecting latest current understanding of cesium speciation, specifically the treatment for Cs_2MoO_4, derived from the Phebus experiments, that differs from MELCOR default values in the last published version of the MELCOR User Guide. In some cases, benchmark analyses are presented in the Best Practices document to document the rationale for the current best practice input values; however, in time, it is intended that such best practice input values will be made default values or documented code user options and described in subsequent versions of the MELCOR User Guide. Many of the recommended benchmark problems can be found in the SAND2001-0929P, MELCOR Computer Code Manuals Version 1.8.5 Volume 3. Chapter 13 of this code manual volume is a comparison to TMI-2. For the ABCOVE comparisons refer to SAND-94-2166, "MELCOR 1.8.2 assessment: Aerosol experiments ABCOVE AB5, AB6, AB7, and LACE LA2." Separate assessments against the Westinghouse 1/7th scale tests have been performed and serve as the basis for current MELCOR counter-current natural circulation treatment, as described in the draft version of the Best Practices document. Current default values used in the MELCOR Larson Miller treatment for lower head creep rupture are taken from the Sandia and OECD Lower Head Failure experiments. It will be verified that these are documented in the MELCOR User Guide. ABCOVE experiments are to be added to the code benchmarks archived in the code manual Volume 3. The VANAM M3 test, very similar to the DEMONA experiments, assessing aerosol depletion for hygroscopic aerosol, is already included in the Volume 3 documentation. Test HDR E11.2 has been assessed under a separate study focused on a comparative evaluation of CONTAIN and MELCOR, along with many other containment experiments including the IET tests, NUPEC Mixing Tests, ISP-41 Iodine Chemistry, HDR V-44, AHMED, Lace LA-4, CSE A9, the NTS H2 Burn Experiments and the GE Suppression Pool Venting Experiments. These containment assessment experiments are planned for inclusion in the MELCOR Volume 3 documentation.

DAVID LEAVER

April 30, 2010 SOARCA Peer Review Draft Report

39. Page C-5 (repeated in December 3, 2010 Peer Review Memo, p. 8)

Comment:

My comment 8a. asked why no mitigated STSBO sequences (i.e., STSBO sequences with 10CFR50.54(hh) measures considered) were included for Peach Bottom. The reason given in comment resolution was that no mitigated STSBO was addressed since the outcome would be the same as mitigated LTSBO (i.e., no core damage). This is not correct. According to the SOARCA analysis, STSBO will cause core damage even if the 50.54 (hh) portable injection is started at 3.5 hours (as it was in the LTSBO). A correct comment resolution should be documented. Also, if it is decided not to include a mitigated STSBO for Peach Bottom, the reason should be provided in the SOARCA report.

Related Comment (December 3, 2010 Memo, p. 7):

Appendix B – Comment 84: The reason given for the my comment on the need to provide a mitigated STSBO for Peach Bottom was that no mitigated STSBO was addressed since the outcome would be the same as mitigated LTSBO (i.e., no core damage). This is not correct. According to the SOARCA analysis, STSBO will cause core damage even if the 50.54 (hh) portable injection is started at 3.5 hours (as it was in the LTSBO). A correct comment resolution should be documented. Also, if it is decided not to include a mitigated STSBO for Peach Bottom, the reason should be provided in the SOARCA report.

Resolution:

The Peach Bottom STSBO proceeds too rapidly to allow for alignment of portable injection; therefore, a mitigated STSBO is not included. It has been judged that alignment of portable injection pumps aimed at mitigating the STSBO cannot be achieved in time to prevent core damage. Therefore, a mitigated STSBO sequence utilizing 10 CFR 50.54 (hh) measures is not performed. However, since RCIC is an existing, in-place, safety system and since RCIC blackstart procedures are in place, we have performed a STSBO calculation assuming RCIC blackstart at one hour. This calculation has been included in Appendix A, Section 5.3, “Short-Term Station Blackout with RCIC Blackstart.” The blackstart of RCIC mitigated the scenario by substantially delaying the onset of core damage. This additional time would allow opportunities for further mitigation (to arrest core heatup) using the portable diesel-driven pumps. Also blackrun of RCIC would allow for further mitigation. An explanatory discussion will be added to Appendix A.

40. Page C-5 (repeated in December 3, 2010 Peer Review Memo, p. 8)

Comment:
There are some places where the operator mitigation strategy assumed (or not) in SOARCA is questionable. One example is comment 1 above. Another is for Surry STSBO where there are reasons why the operator might install portable vessel injection as opposed to installing portable containment spray (operator will not necessarily know if and when lower head failure occurs, so he/she may opt to inject in the hope of preventing lower head failure; he/she may also opt to inject in the hope of minimizing the chance of induced SGTR; finally, the portable injection pump may be able to be installed sooner than 3.5 hours (3.5 hours was assumed for the Surry LTSBO) which would prompt the operator to go down this path as opposed to containment spray. It is suggested that a table and/or text be included in the report that presents the mitigation strategies (50.54(hh)) and the basis for the particular strategy and timing assumed so as to qualitatively strengthen the justification for mitigation success.

Resolution:
The analysis of the mitigative actions was carefully reviewed with plant operations personnel numerous times, especially in the cited STSBO scenario. Through the plant's severe accident management guidelines, there are a number of alternative mitigation options. In discussions with the plant, the range of mitigative actions were identified that would be applicable to the specific scenario and the known conditions. The implementation of the specific actions would be coordinated through the technical support center as these actions fall outside the normal emergency operating procedures. As pointed out by Dr. Leaver, this step requires human evaluation in conditions with little if any instrumentation and perhaps difficult conditions from a seismic event.

Due to the judged difficult accessibility conditions and the lack of alternating current and direct current power, the transient would be more rapid and not allow recovery in-vessel. As pointed out, a TI-SGTR could also be a concern. An extensive mitigative study was not performed but rather an assessment of one action, containment sprays using the emergency Godwin pump. The plant confirmed the preference of this action and its implementation, given the assumption that no external independent pump could be implemented before 8 hours. In a recent fact check comment response from the plant, however, they estimated that it would take approximately 2 to 2.5 hours to ready an external independent pump under normal conditions (based on table-top exercises and timed testing); this will be referenced in the Surry report.

It was judged that, due to difficult accessibility conditions, vessel failure would have occurred before an emergency depressurization using the pressurizer, PORVs, emergency injection into the vessel, or emergency feedwater could be established. Hence, this late mitigative strategy was selected as the best option. The benefits and drawbacks of this strategy were investigated through many MELCOR sensitivity calculations. In particular, the operation of the sprays could lead to large deflagrations and/or detonations. However, the sprays also cooled the ex-vessel debris and knocked down the airborne fission products. Consequently, the mitigative strategy showed net benefits.

In SOARCA, the mitigative selection exercise was done with assistance of plant personnel. The selected actions were based on the best option for the specified conditions. The relative likelihood of selecting the less likely actions was not assessed. However, it should be noted that the failure to successfully implement a mitigative action was examined in the unmitigated case. A probabilistic evaluation would consider a range of strategies and evaluate their relative likelihood and consequences. However, for the best-estimate purpose of SOARCA, only the most likely option was examined.

Text has been added to the report which discusses NRC staff holding table-top exercises with the licensee to strengthen the basis and timing for particular strategies.

41. Page C-5 (repeated in December 3, 2010 Peer Review Memo, p. 8)

Comment:
Appendix A, Figure 85, 20 mile risk (STSBO with RCIC blackstart) is higher than Figure 87, 20 mile risk (STSBO with no RCIC). At other distances it is the other way around (which is intuitively the way it should be. i.e., with RCIC blackstart, the risks are lower). This should be explained in the text.

Resolution:
There was an error in the calculations, and these figures and text have all been replaced because the source terms were recalculated and, as a result, the consequences were

recalculated. The trend that triggered this concern is no longer present in the current results; the 20-mile risk now follows the other distances, i.e., the risk for STSBO with no RCIC is higher than the STSBO with RCIC blackstart (see Appendix A Section 7.3.2, “Short-Term Station Blackout with Reactor Core Isolation Cooling Blackstart”).

42. Page C-5 (repeated in December 3, 2010 Peer Review Memo, p. 8)

Comment:
Page 68 of the Summary report still says that risks are calculated to 100 miles.

Resolution:
This has been corrected in the Main Report. Section 5.8 Risk Metrics Reported states 0 to 50 miles for the mean population weighted risks.

43. Page C-5 (repeated in December 3, 2010 Peer Review Memo, p. 8)

Comment:
It is suggested that the fifth bullet in the conclusions on page xxix of the Executive Summary be generalized to apply to all sequences that were screened as opposed to just bypass sequences. For example: “Scenarios which are lower frequency than the scenarios which survived the screening criteria would not pose a higher latent cancer fatality risk than the scenarios which survived the criteria since the higher conditional risk is offset by the lower frequency.”

Resolution:
The SOARCA team agrees with the comment in general. This principle is cited in connection with our discussion of the appropriateness of the screening criteria, which were also supplemented by other considerations such as a comparison with important scenarios in past studies such as NUREG-1150. Further, the results of our analyses have confirmed the principle for the other scenarios analyzed. Nonetheless, generalization of the principle to cover all probabilistic considerations, while appealing, is beyond the scope of the study.

44. Page C-5 (repeated in December 3, 2010 Peer Review Memo, p. 8)

Comment:
Suggest changing middle sentence of large paragraph on page 10 of Summary report as follows: “While it is judged, on the basis of the procedures and training, that these measures are expected to be effective, a limitation of this approach is that a comprehensive human reliability

assessment has not been performed to quantify the probabilities of plant personnel succeeding in implementing these measures." QED

Resolution:
The Main Report Section 1.7, "Mitigated and Unmitigated Cases" currently has this statement to discuss the lack of a comprehensive human reliability analysis in SOARCA:

"A limitation of this approach is that a comprehensive human reliability assessment has not been performed to quantify the probabilities of plant personnel succeeding in implementing these measures. However, NRC has issued 10 CFR 50.54(hh) requiring plant licensees to possess the equipment, develop the strategies, and have trained personnel to implement these mitigative measures. The 10 CFR 50.54(hh) measures are the result of a major effort by industry and NRC in the 2004–2008 timeframe to develop means to mitigate events involving loss of large areas of the plant due to fire and explosions.

The assessment of mitigation measures has continued to receive attention since our initial assessment conducted with plant staff. NRC staff performed follow-up site visits in June and August 2010 to explicitly address RCIC blackstart and run for STSBO and manual operation of TD-AFW. The site visits included a review of RCIC blackstart and run procedures, additional tabletop exercises to address conservatism in the assumed PWR STSBO timeline, and plant walkdowns of equipment areas and detailed reviews of procedures. SOARCA staff concluded, following these site visits, a greater likelihood of implementing mitigation." In addition, the recent post-Fukushima inspections at Peach Bottom and Surry provide some additional information with respect to the two plants' abilities to implement SAMGs and B.5.b measures (the inspection reports are available at: http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/follow-up-rpts.html). The SOARCA team is currently evaluating the information in these inspection reports for any relevant insights.

Note that a sensitivity calculation is provided (the "unmitigated" scenarios), to show what the consequences would be without mitigation, which bounds the spectrum of possible consequences with regard to assigning success probabilities to the 10 CFR 50.54(hh) mitigation actions. See further discussion in response to Comment 49 and Mr. Mrowca's Comments 60-61.

45. Page C-5 (repeated in December 3, 2010 Peer Review Memo, p. 8)

Comment:

Summary report, page 22: fourth bullet, frequency range is 1E-7 to 5E-7, not 8E-7.

Resolution:

This has been corrected in the report.

46. Page C-5 (repeated in December 3, 2010 Peer Review Memo, p. 8)

Comment:

Appendix A, Section 5.5 Loss of Vital AC Bus E-12 is titled "Mitigated Response", but it is actually unmitigated per Section 3.3.3 and 3.3.4.

Resolution:

The section titles have been corrected.

47. Page C-6 (repeated in December 3, 2010 Peer Review Memo, p. 9)

Comment:

My comment 2 in the August 5, 2009 comment set suggested benchmarking MELCOR against the TMI-2 accident. The comment response said validation against TMI-2 would be of limited benefit considering the accident sequences of interest to SOARCA. I think this resolution misses the point. The TMI-2 accident is a very important and useful benchmark on core damage progression and fission product release to the primary system and containment (which in turn determine much of what happens later in time in the accident), and it would be a good idea to benchmark the revised MELCOR model (e.g., hemispherical lower head) against TMI-2. This may not be practical as part of SOARCA, but it would be valuable and should be considered longer term.

Related Comment (December 3, 2010 Memo, p. 7)**:**

Appendix B – Comment 28: The resolution to my comment on the need for TMI-2 benchmarking for MELCOR misses the point. The resolution stated that, "Validation against the TMI-2 event which had a very limited release would also be of limited benefit considering the accident sequences of interest to the SOARCA project." The TMI-2 accident is a very important and useful benchmark on core damage progression and fission product release to the primary system and containment (which in turn determine much of what happens later in time in the accident), and it would be a good idea to benchmark the revised MELCOR model (e.g., hemispherical lower head) against TMI-2. This may not be practical as part of SOARCA, but it would be valuable and should be considered longer term.

Resolution:

The original comment resolution indicated validation against TMI-2 would be of limited benefit considering the accident sequences of interest to SOARCA. The SOARCA team reconsidered your comments and agrees that the TMI-2 accident is indeed a very important and useful benchmark. The MELCOR Code Manual Volume 3 (SAND2001-0929P) contains a collection of benchmark and validation analyses with MELCOR Version 1.8.5. Central values of key MELCOR parameters are derived from the assessments in MELCOR Volume 3. Chapter 13 of this code manual volume is a comparison to TMI-2. (See also the resolution to Dr. Henry's Comments 34 and 38.)

48. Page C-6 (repeated in December 3, 2010 Peer Review Memo, p. 9)

Comment:

The resolution to Comment 49 on the Summary document says that a short paragraph was inserted in the Executive Summary to describe the fraction of emergency phase risk within 10 miles that is attributed to the non-evacuating cohort. I could not find such a paragraph in the Executive Summary.

Resolution:

Consequence results for the nonevacuating cohort will continue to be included in the overall consequence calculations and a short discussion of the nonevacuating cohort will be added to the report. In some of the slowly developing sequences, 100% of the emergency phase risk is from nonevacuees.

49. Page C-6 (repeated in December 3, 2010 Peer Review Memo, p. 9)

Comment:

The resolution to Comment 85 on the Summary document says that the Executive Summary has been enhanced to emphasize that the probability of 50.54(hh) mitigation is assumed to be zero for purposes of unmitigated sequences. I could not find this in the Executive Summary.

Previous Comment 85:

For the same reasons as described in my August 5, 2009 Comment 5, some reasonable probability should be assigned to operator failure to implement the 50.54(hh) mitigative measures. If a factor of 10 is assumed as was done in the August 5, 2009 Comment 5, the unmitigated STSBO sequences (two of them) probabilities would decrease to 1E-8 – 5E-8, and the mitigated STSBO sequences (if they were added to the analysis) would be 1E-7 – 5E-7. (cf. detailed comments by Leaver 10/5/09 for frequency estimates)

Resolution:

The inclusion of both mitigated and unmitigated results is an important feature of the SOARCA results and demonstrates the impact of potential mitigation. While addressing this comment would demonstrate, in a more integrated risk sense, the impact of the 10CFR 50.54(hh) measures, the effort would require a risk and human reliability study, which is beyond the scope of the SOARCA project. A probability/frequency is not assigned to mitigation. Rather a sensitivity calculation is provided to show what the consequences would be without mitigation (the “unmitigated” scenarios), which bounds the spectrum of possible consequences with regard to assigning success probabilities to the 10 CFR 50.54(hh) mitigation actions. The Executive Summary has been enhanced to emphasize that the probability of 10CFR50.54(hh) mitigation is assumed to be zero for the purposes of the SOARCA analysis of the unmitigated cases.

50. Page C-6

Comment:

The Appendix B, page 174 footnote states that inertial deposition is expected to be a significant capture mechanism in the LHSI line, and that other mechanisms “were important”. Is “were important” a typo?

Resolution:

The analysis and text describing the ISLOCA are currently being revised as a result of fact check comments received from the licensee in April 2010. It is expected that the new analysis will address both turbulent deposition and inertial deposition (impaction).

December 3, 2010 Peer Review Memorandum

51. Page 7

Comment:
Best Modeling Practices – Comment 11: The resolution to my comment on the term “physically unreasonable” notes that this phrase “has become a term of art” for remote probability or essentially impossible in an LWR severe accident environment (my words). I still believe that “physically unreasonable” will not connote remote probability or essentially impossible to many that will read the SOARCA document, and I urge the staff to provide clarification in the text where this term is used.

Resolution:

This term is generally associated with events that are deemed so infrequent as to not warrant quantification. It is a term that evolved from historical research into steam explosions, and drywell liner melt through.

The SOARCA team reviewed the use of the term, “physically unreasonable,” in the SOARCA reports and the team thinks that adequate clarification is provided to convey the intended meaning where the term is used. For example, note the words emphasized in the following quote from Section 4.8.2, “Early Containment Failure Phenomena,” in Appendix A: “The issue of Mark I drywell shell (liner) melt-through at Peach Bottom was assessed by the NUREG-1150 molten core-containment interaction panel. The results of expert panel elicitation are reported in NUREG/CR-4551, Volume 2, Revision 1, Part 2, “Evaluation of Severe Accident Risks: Quantification of Major Input Parameters, Experts' Determination of Containment Loads and Molten Core Containment Interaction Issues,” issued April 1991. The response was uncertain because there were two schools of thought on this issue. Since the completion of NUREG-1150, the NRC has sponsored analytical and experimental programs to address and resolve this “Mark I liner attack” issue. The results of an assessment of the probability of Mark I containment failure by melt attack of the liner were published in NUREG/CR-5423, “The Probability of Liner Failure in a Mark I Containment,” issued in 1989 and NUREG/CR-6025, “The Probability of Mark I Failure by Melt-Attack of the Liner,” issued in 1993. It was concluded that, in the presence of water, <u>the probability</u> of early containment failure by melt-attack of the liner <u>is so low as to be considered physically unreasonable</u>.” (emphasis added)

BRUCE MROWCA

April 30, 2010 SOARCA Peer Review Draft Report

52. Page 35

Comment:

Recommendation 1: Provide a summary of the changes that are being incorporated in response to the first supporting objective. Consider rewording this objective to reflect a balanced consideration of risk-significant improvements and challenges.

Associated discussion (p.32): …the stated objectives also appear to be positively biased as indicated by the framing of the first supporting objective. This objective addresses "improvements and updates" as opposed to changes not reflected in earlier assessments. Although this reviewer agrees that there have been many improvements, there are also significant challenges associated with areas such as new fire model methods, increased on-line maintenance or generic issues such as the sump screen issue. A balance discussion should be included in the SOARCA report of the method used to select the changes for incorporation into this project in order to inform the reader as to potential limitations that may not have been addressed.

Resolution:

Improvements in fidelity and modeled phenomena or state of knowledge are not presumed to lead to more favorable predictions of risk. Rather, improvements are assumed to lead to more accurate or realistic predictions. Modeling refinements are neutral with respect to outcome and include effects that can lead to either larger or smaller releases. The SOARCA study was not biased to reflect only effects that would reduce predicted source terms. For example, plant representations that include higher burnup fuel inventories likely lead to larger absolute releases (the second supporting objective mentions this as, "Incorporation of plant changes, such as power uprates and higher core burnup, that are not reflected in earlier assessments;…"; see also the response to Dr. Henry's Comment 31). Good accounting of deposition surfaces could lead to lower absolute releases. Overall, the intent is to render more realistic predictions which the SOARCA team deems a balanced approach.

The SOARCA team attempted to accurately reflect plant conditions. The latest Level 1 information concerning the initiators derived from the plant-specific SPAR models has been used and these are not believed to be biased. Additionally, realistic information such as fuel burnups, power uprates and contemporary higher population densities, all of which have the effect of increasing negative consesequences, are accounted for as well in this intended best-estimate treatment.

53. Page 35

Comment:

Recommendation 2: Provide a better justification for the selected screening criteria.
Associated discussion (p. 33): The case for using the selected screening process is not well made. The analysis states that the priority of the work is to bring a "more detailed, best-estimate, and consistent analytical modeling to bear in determining realistic outcomes of severe accident scenarios" and concludes that the benefits could most efficiently be demonstrated by applying these methods to a set of the more important severe accident sequences. However, the stated project objectives are much farther reaching than demonstrating the benefits of realistic analytical methods. The benefits of realistic analysis can be achieved by selecting any relevant set of sequences. For the narrow objective of demonstrating the benefits of realistic methods, this reviewer agrees that approach taken is sufficient. However, the other identified objectives suggest that it is necessary to capture all or a significant portion of the risk. Specifically, a more comprehensive approach would appear to be called for in order to communicate risk and to provide an update of the quantification of offsite consequences contained in NUREG/CR-2239.

Resolution:

Section 2.1, "Approach," of the Main Report discusses the approach the SOARCA team used, which considered accident scenarios (sequence groups, rather than individual sequences) based on both likelihood and potential consequences. Core-damage sequences from previous NRC and licensee PRAs were identified and binned into core-damage groups. A core-damage group consists of core-damage sequences that have similar timing for important severe accident phenomena and similar containment or engineered safety feature operability. The *groups* (not individual sequences) were screened according to their approximate core-damage frequencies to identify the most significant groups. Since core-damage groups, i.e., scenarios, were considered, many individual lower-order sequences would be captured in the aggregation into groups.

The SOARCA team used selection criteria based on core damage frequency (based on the latest NRC SPAR models) and initiating events involving containment bypass or leading to an early failure of the containment with potential consequences. Scenarios (sequence groups) with a frequency at or above 10^{-6} per reactor-year were screened in, as well as scenarios with a frequency at or above 10^{-7} per reactor year for scenarios that are known to have the potential for higher consequences (e.g., containment bypass scenarios such as steam generator tube rupture and interfacing system loss-of-coolant accident [ISLOCA] initiators)

The selected scenarios for SOARCA were shown to be important in recent and past probabilistic assessments, particularly for the well-studied pilot plants of Peach Bottom and Surry. The selected scenarios are also representative of broad classes of transients. The selection of station blackout (SBO) events in SOARCA ensures that we have covered that broader class of transients involving a loss-of-heat removal and further, by including a short-term blackout we have reasonably bounded that class of accidents (which could include other events such as loss of service water or loss of component cooling water but which develop more slowly). Also, for the PWR, the station blackout also includes, in part, the effect of a small loss of coolant by considering reactor coolant pump seal leakage. In addition, by the selection of station blackout sequences for analysis, we also include the effects of loss-of-containment heat removal (fan coolers) and loss-of-containment spray systems (which are all electrically powered) to remove airborne radionuclides.

The purpose of SOARCA was to do a best estimate consequence analysis. Pursuing a Level 3 PRA which would capture consequences from additional lower order sequences would necessitate introducing conservative assumptions in order to complete analyses and would have served a different purpose.

In addition, the most recent Level-3 PRA-type information we have, while somewhat simplistic and approximate, does indicate that SOARCA is likely to have considered accident scenarios that account for the majority of risk at both the Peach Bottom and Surry plants. The Severe Accident Mitigation Alternatives (SAMA) analysis submitted with the license renewal application (2001) for Peach Bottom indicated that station blackout scenarios accounted for the majority off-site risk (Reference: Peach Bottom, License Renewal Application, Environmental Report, Appendix G, available at:
http://www.nrc.gov/reactors/operating/licensing/renewal/applications/peach-

bottom/peach_bottom-envg.pdf). The SAMA analysis for the Surry license renewal application (2001) indicated that SGTR and ISLOCA accounted for the majority of the off-site population dose risk (Reference: Chapter 5 of NUREG-1437, Supplement 6 (2002),), "Generic Environmental Impact Statement for License Renewal of Nuclear Plants: Regarding Surry Power Station, Units 1 and 2 – Final Report," available at: http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1437/supplement6/2671bpart2.pdf).

Additional analyses and discussion were added to Chapter 7 of Appendix A to include a much more detailed comparison of NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," to SOARCA, and this will be presented at the upcoming planned Peer Review Committee meeting.

54. Page 35

Comment:

Recommendation 3: Provide the risk profile that is being assumed for the assessment of each plant. Although it is understood that this profile is estimated and is developed based on multiple models, it is impossible to judge the degree of risk being capture by this analysis without a clear starting point.

Associated discussion (p. 34): This reviewer believes that the targeted sequences identified in the SOARCA report represent significantly less than the 95% ASME PRA criterion… to obtain the identified screening criteria would require a significantly lower screening value, at least one order of magnitude lower, than that used by the SOARCA Project. The use of the acceptable surrogate goal for the quantitative health objectives contained in the Commission's Safety Goal Policy statement as opposed to the estimated CDF associated with each plant, likely results in significant risk being screened.

Resolution:

SOARCA is not presuming any initial risk profile, nor does it seek to quantify change in plant risk (the resolution to Comment 53 above discusses the scenario selection process). The SOARCA study is not a traditional PRA which is intended to quantify total risk from all sequences, and hence the American Society of Mechanical Engineers PRA criterion was not applied. However, as noted in the resolution to Comment 53 above, the SAMA analyses contained in the 2001 license renewal applications and associated final supplemental environmental impact statements indicate that SBO scenarios account for the majority of offsite risk for Peach Bottom, and ISLOCA and SGTR scenarios account for the majority of the offsite population dose risk at Surry.

55. Page 35

Comment:

Recommendation 4: Ensure that the presentation of accident sequences is consistent between the executive summary and the appendices.

Associated discussion (p. 34): The SOARCA Executive Summary shows that four accident sequences were selected for Surry's consequence analysis with three identified as external event related and one identified from the internal events PRA. The total frequency of these events is 2×10^{-5}. Appendix B contains some variations to this list including an additional internal sequence associated with a spontaneous steam generator tube rupture (SGTR).

Resolution:

The Executive Summary has been revised, and the discussion of Surry sequences should now be consistent.

56. Page 35 (SOARCA peer review report)

Comment:

Recommendation 5: Provide explicit mapping of the sequences from the set of initial sequences for those that met the screening criteria to those that were considered in the consequence analysis. Ensure that the frequency for each sequence is explicitly identified. Ensure that the reason for elimination of a sequence is clearly stated.

Recommendation 6: Account for all significant sequences.

Associated discussion (p. 34): A review of the internal event sequences contained in the Surry SPAR Model shows that the two internal event sequences selected for the SOARCA Project represent less than 15% of the internal events contribution to core damage and that depending on the approach used to bin the accident sequences several other sequences may have candidates for inclusion in the consequence analysis even if the 10^{-6} criterion was used … Some of these sequences may be considered to have been bounded by the long-term and short-term station blackout (SBO) scenarios, but as currently written, these blackout scenarios appear to be addressing external event challenges and are separate from the internal event-related sequences.

Resolution:

Additional text and references have been added to the reports about the scenario selection process. The scope of the SOARCA project in terms of number of accident sequences was intended to be representative, and was intentionally not organized to be a comprehensive risk study.

The SOARCA team contends that the selected sequences are representative in terms of reflecting the range of potential consequences. We feel that the scenario selection process for SOARCA assures that the more important severe accident scenarios are captured in terms of consequences (especially with the addition of scenarios that would have been screened out based on frequency but included due to historic interest), even though the study is not a traditional PRA. Modeling the consequences of SBO scenarios initiated by external events (e.g., seismic) should be bounding for the consequences from internally-initiated SBO events, since the externally-initiated events assumes the unavailability of additional equipment disabled due to the initiating event.

57. Page 36

Comment:

Recommendation 7: Define the sequence framework that is being used in the SOARCA Project. Ensure that it is consistent with the screening criteria.

Resolution:

As described previously, the SPAR models for the subject plants were used to identify candidate scenarios in the SOARCA framework. The scenario selection process is discussed in response to Comment 53 above.

58. Page 36

Comment:

Recommendation 8: Provide a summary table within each appendix that identifies each sequence meeting the screening criteria, and its treatment within the accident progression and the emergency response sections. Give each sequence a unique identifier and address it in the same order within each section.

Resolution:

Section 2 of the Main Report describes the scenario selection process in detail. Short discussions are also being added to the Appendices. The number of scenarios in the SOARCA project is not extensive; the SOARCA team does not see the need to assign each scenario a unique identifier. The Main Report and Appendices are being reviewed to ensure that scenarios are addressed in a logical and consistent order.

59. Page 36

Comment:

Recommendation 9: Include the identification and/or development of each sequence frequency within Section 3 of each appendix.

Resolution:

The accident scenarios (sequence groups) were developed as an initial step in the project following criteria that was applicable to both plants. The information on the accident scenarios is presented in Section 2, “Accident Scenario Selection,” of the Main Report. As explained there, scenario frequencies were based on the latest NRC SPAR models and discussed with each licensee. Further work quantifying frequencies was not part of the SOARCA project.

60. Page 36

Comment:

Recommendation 10: Perform a human reliability assessment for the identified security-related mitigation improvements or identify a conservative screening value so that all sequence frequencies can be calculated.

Resolution:

SOARCA does not include a quantitative human reliability analysis; however, the 10 CFR 50.54(hh) procedures and training were inspected as part of security assessments with site-specific evaluations prepared. Additionally, SOARCA staff performed follow-up site visits in June and August 2010 and January 2011 to explicitly address Reactor Core Isolation Cooling (RCIC) blackstart and run for STSBO and manual operation of TD-AFW and to discuss fact check comments. The site visits included a review of RCIC blackstart and run procedures, additional tabletop exercises to address conservatism in the assumed PWR STSBO timeline, and walkdowns and detailed reviews of procedures (see discussion in Appendix A Section 2.3

and Appendix B Section 1.5). In addition, the recent post-Fukushima inspections at Peach Bottom and Surry provide some additional information with respect to the two plants' abilities to implement SAMGs and B.5.b measures (the inspection reports are available at: http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/follow-up-rpts.html). The SOARCA team is currently evaluating the information in these inspection reports for any relevant insights.

A probability/frequency is not assigned to mitigation. Rather, a sensitivity calculation is provided to show what the consequences would be without mitigation, which bounds the spectrum of possible consequences with regard to assigning success probabilities to the 10 CFR 50.54(hh) mitigation actions. Further, this "unmitigated" scenario specific risk quantification assumes the probability of 10CFR50.54(hh) mitigation is zero, which is conservative and bounding. See also resolution to Comment 61.

61. Page 35

Comment:

Conclusions: It is clear that the insights gained from the integrated phenomenological analysis using self-consistent scenarios are significant and the report demonstrates the benefits of this more realistic treatment. However, the focus of this review was on the process for selecting the scenarios and on applying the security-related recovery actions. These activities appear to have serious limitations. The scope of changes considered by SOARCA was not clearly stated, the starting risk profiles of the selected plants was not provided, the appropriateness of the sequence screening criteria was not well defended, the calculation of the sequence frequencies was incomplete and a state-of-the-art human reliability analysis of the security-related actions was not performed. These weaknesses reduce the confidence that all of the stated objectives were met.

Resolution:

SOARCA is demonstrating new approaches to evaluating consequences of severe accidents and is not intended to be a full scope PRA. The intent of SOARCA was to provide a more realistic consequence assessment of traditionally studied accident sequences that are important; it was not the intent to quantify any change in baseline risk of the plants. The SPAR process and the applied screening criteria produced a spectrum of accidents that are similar to (or consistent with) the severe accidents traditionally used to inform source term categories in PRAs. One important feature of the SOARCA approach is that the sequences are examined in significant detail for realism and not subjected to simplifications inherent in traditional PRA. Realism was emphasized over conservatism.

To the extent possible, human actions were considered, especially with regard to mitigation actions, and when it was considered that insufficient time was available to implement mitigations, they were not credited. The project did discuss the state of human factors sciences and mitigations for which there are no currently well developed procedures. It was a pragmatic decision to explore mitigation actions nevertheless, considering that they would be attempted. A probability/frequency is not assigned to mitigation, rather, a sensitivity calculation is provided to show what the consequences would be without mitigation, which bounds the spectrum of possible consequences with regard to assigning success probabilities to the 10 CFR 50.54(hh) mitigation actions. As noted for Comment 60, the quantification assumed the likelihood of 10CFR50.54(hh) mitigation was zero.

While SOARCA does not include a quantitative human reliability analysis, the 10 CFR 50.54(hh) procedures and training were inspected as part of security assessments with site-specific evaluations prepared. Additionally, SOARCA staff performed follow-up site visits in June and August 2010 and January 2011 to explicitly address licensee fact check comments. The site visits included a review of RCIC blackstart and run procedures, additional tabletop exercises to address conservatism in the assumed PWR STSBO timeline, and walkdowns and detailed reviews of procedures (see discussion in Appendix A Section 2.3 and Appendix B Section 1.5). In addition, the recent post-Fukushima inspections at Peach Bottom and Surry provide some additional information with respect to the two plants' abilities to implement SAMGs and B.5.b measures (the inspection reports are available at: http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/follow-up-rpts.html). The SOARCA team is currently evaluating the information in these inspection reports for any relevant insights.

Future PRA could be evolved to include such features and future reliability studies performed to establish success frequencies for risk quantification purposes.

KEVIN O'KULA

Comments dated May 17, 2010

62. Page 3

Comment:

While satisfactory MELCOR- WinMACCS integration was apparently achieved, much of this work was not documented to the appropriate level of detail that would be desirable in a study of this magnitude. More has been said about the integration in Section 6. However, the chronological treatment applied in the SOARCA analysis was notably consistent from scenario selection through offsite consequence evaluation for each of the eight baseline, accident sequences discussed in the NUREG report.

Resolution:

A separate document to describe the MACCS2 methods and choices used in SOARCA is planned (MACCS2 best practices for SOARCA) and should resolve this comment. This document will be published at a later date (a draft is expected to be available in late 2012).

63. Page 3

Comment:

In general, the overall technical results are well substantiated and explained in sufficient detail so as to support key findings and study insights. While good use is made of the NUREG/CR-2239 (Ref. 3) SST1 source term with respect to the composition, timing, and magnitude of the release relative to SOARCA source terms, the opportunity should be seized to connect with Peach Bottom and Surry results from NUREG-1150 where practicable. The SOARCA study is an opportunity to build on the discussion from the landmark severe accident risk study for Surry and Peach Bottom to show how improvements in methods, training, modeling, plant improvements, have substantially reduced severe accident risks. This comparison would be highly informative for those accident sequences, e.g., station blackout or LOCA that were analyzed throughout the 1970s - 1980s and have been revisited during the SOARCA study.

Resolution:

As the comment indicates, we currently give no comparisons with NUREG-1150, at least for the consequence results. While the SOARCA team sees the benefit of this recommendation, comparisons would be difficult. Source terms in NUREG-1150, "Severe Accident Risks: An

Assessment for Five U.S. Nuclear Power Plants," were grouped into source term categories in a way that makes it impossible to trace them back to the initiating event. For this reason, direct comparisons with NUREG-1150 have not been attempted.

As part of Appendix A (Section 7.3.6, "Comparison with Sandia Siting Study"), additional analyses and discussion were added to the report to directly compare the results from NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," to SOARCA. Specifically, since the 1982 Siting Study does not provide latent cancer results at distances that are meaningful and comparable to those provided in the SOARCA study or to the NRC safety goal, an effort was made to reproduce the 1982 Siting Study results for Peach Bottom using the SST1 source term in order to produce results that are directly comparable to the SOARCA results. A comparable write-up will be added to Appendix B for Surry.

The purpose of SOARCA is to do a best-estimate analysis on analyzed scenarios. The overall objective is to develop a body of knowledge regarding the realistic outcomes of severe reactor accidents. One supporting objective is to incorporate the significant plant improvements and updates not reflected in earlier assessments, and another supporting objective is to evaluate the potential benefits of recent security-related mitigation improvements in preventing core damage and reducing an offsite release should one occur. SOARCA is not intended, however, to quantify the risk reduction from improvements.

64. Page 5 (Repeated in December 3, 2010 Peer Review Memo, p.10)

Comment:

MELCOR-to-MACCS2 transition - The documentation in the four-volume NUREG report, and especially in Volume I, Summary report, is sparse with respect to the MELCOR to MACCS2 transition. It is difficult to judge how best-estimate aspects of the source term description are based the description provided for deposition velocity and surface roughness length. In the Summary report, the discussion (pages 60-61) is ambiguous regarding the approach to assigning deposition velocity to aerosol particle sizes. Specifically, it is unclear if binning associated with particle size and deposition velocity uses expert elicitation and MELMACCS methodology or if one approach was primary and the other supplementary. The pedigree of the MELMACCS technical report (Ref. 47) appears to be at an internal laboratory report. It is recommended that the report be formally released with adequate technical review.

Resolution:

Additional text has been added to Section 5.4, “Source Term Evaluation,” of the Main Report which discusses the MELMACCS treatment of deposition velocity and aerosol particle sizes.

Providing an explanation for the connection between MELCOR and WinMACCS would be beneficial for MACCS2 modelers who would like to update the way they import source term data from the methods used at the time of NUREG-1150, which is still common practice among MACCS2 modelers. A general description of the connection between MELCOR and WinMACCS will be added to the MACCS2 best practices for SOARCA document, which will be published at a later date (a draft is expected to be available in late 2012).

65. Page 5

Comment:

Surface roughness length (z_o) - The documentation in the Volume I Summary is very brief and omits much technical justification for selection of key parameters. One of the areas that remains too limited is the basis for the surface roughness length (z_o). Ten centimeters surface roughness length would seem to be overly conservative for Peach Bottom given the fifty-mile environment surrounding the plant. The same value may be appropriate for Surry but a more complete discussion is recommended. Is the 10-cm value used for both plants an indication that the environments around the plants were considered characteristic of tall grass (refer to Table 2.3 from NUREG-4691 Vol. 2 and shown in Figure 1 below)?

Resolution:

The 10 centimeter (cm) value for surface roughness has been used as a typical value for the US and site-specific values have generally not been considered in previous analyses, such as NUREG-1150. To address the choice of surface roughness, a brief discussion of the selection of the 10-cm roughness length has been added to Sections 5.4, “Source Term Evaluation,” and 5.5, “Site-Specific Parameters,” of the main document. Also, a sensitivity study varying surface roughness is currently being performed, and we will include the results in the SOARCA best-estimate report Appendices. The sections we are adding to discuss these sensitivity analyses will provide a more detailed discussion of the selection of site-specific surface roughness for Peach Bottom and Surry. We also plan on addressing this during the uncertainty analysis for the project. See additional discussion in resolution for Comment 66.

66. Page 6

Comment:

Deposition velocity selection - The deposition velocities associated with various aerosol sizes is given in Table 13 of the Summary (page 61). However, it is not clear what radionuclides groups are associated with one or more of the bins shown in Table 13 and how the median diameter bins would be distributed for a given radionuclide group. Surely a state-of-the-art input distribution of deposition velocities would offer a different distribution based on physicochemical characteristics of one group vs. those of another.
For example, the halogen isotopes (primarily radioiodine) will deposit differently from cesium species (e.g., CsI and CsMo04). It is suggested that while realistic input is reflected in the SOARCA study and used in the current analyses, justification should be provided to support its use in place of the sample input published in the 1998 MACCS2 User's Guide (Ref. 5) for the allocation fractions for nine fission product groups (Figure 2).

Related Comment (December 3 Memo, p. 11)**:**

It is suggested that while realistic input is reflected in the SOARCA study and used in the current analyses, justification should be provided. A draft copy of Ref. 48 (SOARCA Summary document) seems to indicate that deposition velocity was based on an expert-elicitation approach (page 43), with prairie, forest, and urban surface roughness length used as a parameter by the experts. The overall process that was applied is not apparent and it would be greatly benefit the intended NUREG documentation if additional detail could be provided.

Furthermore, I am apparently in the minority with the approach used but believe it paints a picture that deposition would occur in the same way regardless of the site of interest. Surry and Peach Bottom may be sufficiently alike to justify this approach for SOARCA, but in general I don’t believe it is correct for smaller size aerosols that interact with the surface features, and as such, are not controlled by settling velocity alone.

Resolution:

Section 5.4, “Source Term Evaluation,” of the Main Report describes how typical values for surface roughness and mean wind speed, 0.1 m and 2.2 m/s, respectively, are used as inputs to MELMACCS in order to calculate deposition velocity. Mean wind speeds were determined from the specific weather files used in the consequence analyses. Table 3 of the Main Report displays the deposition velocities used in SOARCA analyses for both Peach Bottom and Surry.

In the MELCOR treatment of aerosols the size distribution is discretized into size intervals or sections based on the "sectional treatment" of the MAEROS model. Within each size section, all particles are identical and each particle is considered to be composed of proportional fractions of the chemical groups present as aerosols in a particular volume. Micro-photographs show particles are agglomerations of many small particles. Even the smallest of these, which are sub-micron are comprised of many elemental forms. Since they are formed as condensation nucleii from co-mixed vapors, there really isn't any way for them to partition into distinct pure particle species. Since most of MELCOR aerosol size bins are populated as fission product vapors are condensing and particle number densities are very large, agglomeration rates are therefore very large. In that case, there is no way to keep them apart and it is a very good approximation.

At the same time, the fractional composition of the chemical classes within each size section varies over the transient and the aerosol size distribution itself also changes over the course of the transient. Because of this, the average (over the time of the accident) particle size distribution is different for each chemical class. Using the transient information in the MELCOR plot file, MELMACCS calculates a particle size distribution for each chemical class using the MELCOR sectional class fraction information, and uses that to define an independent size distribution for each chemical class.

A brief description of the calculation of aerosol size distributions is now included in Section 5.4 of the Main Report. An expanded description is planned to be included in a separate document that describes SOARCA modeling choices and methods for consequence analysis (MACCS2 best practices for SOARCA), which will be published at a later date (a draft is expected to be available in late 2012).

MACCS2 uses the aerosol size distributions by chemical group, plus the deposition velocities in the table, to determine the rate of depletion of aerosols from the plume. Generally, the larger aerosols deposit more quickly and so are depleted more rapidly from the plume. The peak in the aerosol size distribution is usually a few micrometers, which corresponds to a deposition velocity of a few millimeters per second.

The sensitivity study to explore the effect of surface roughness, mentioned in the above response to Comment 65, will also include the effect of surface roughness on deposition velocity. So, the study will include the effects of surface roughness on vertical dispersion and on deposition velocity for both the Peach Bottom and Surry sites. The results of the sensitivity study will be folded into Chapter 7 of Appendices A and B.

67. Page 6 (repeated in December 3 memo, p. 11)

Comment:

Discussion of the non-site specific and site-specific parameters used: A key outcome of the SOARCA project, discussed several times with the Peer Review Panel during our meetings, has been the smaller and delayed source term compared to the Sandia Siting Study. These important results are illustrated in Table 21 (Appendix A – Peach Bottom Integrated Analysis) and Table 32 (Appendix B - Surry Integrated Analysis). Especially in the case of the Surry conditional risks with distance (Figures 172, 174, 176, 178, and 180), the results are dominated by the long-term (CHRONC) phase of the analysis. This aspect of the model is driven by user input of dose criteria (habitability dose limits), site-specific data, and some non-site specific information. It would improve the understanding of the SOARCA study to provide the tables of information to augment the discussion of sections 5.5 and 5.6. The tables would mostly address the type of inputs that control the EARLY and CHRONC modules, specifically in deciding on condemnation and inhabitation return and be as follows (with suggested location by volume):

Table 1. Peach Bottom Offsite Consequence Analysis Site-Specific Parameters (Appendix A)
Table 2. Surry Offsite Consequence Analysis Site-Specific Parameters (Appendix B)
Table 3. Site-Independent Parameters (Summary – Section 5.6).

While some of these data (e.g. shielding factors) can be found elsewhere in the four-volume set of SOARCA documents, it would assist the reader to see the key inputs collected in one or several tables. This information would be useful in determining what sensitivity studies are important, what the focus should be on plant features, severe accident mitigation procedures, etc., and to what extent public policy has a role.

Resolution:

The recommended tables for the input parameters will be added in the SOARCA reports.

68. Page 7

Comment:

Boundary weather – During one of the review meetings, it was indicated whether a boundary weather condition is imposed, with forced deposition conditions, and if so, the type of weather specified and at what region in the grid. I don't think this is established in the February final draft for review document. This aspect of the offsite consequence analysis should be described for comparison with past work (e.g., NUREG-1150).

Resolution:

A brief discussion on boundary weather will be added in Section 5.5, "Site-Specific Parameters," of the Main Report. In all of the SOARCA calculations, boundary weather is applied beyond the 50-mile region for which risk is characterized. Thus, the conditions chosen for boundary weather have no influence on the reported results.

69. Page 7 (repeated in December 3 memo, p. 11)

Comment:

Centralized discussion of MACCS2 improvements- There are many improvements noted in the MACCS2 model with 64 directional sectors and more realistic evacuation modeling among others. It would be informative to have a short section in Appendix A and Appendix B to summarize the prominent features and expanded capabilities by module, i.e., ATMOS, EARLY, and CHRONC. This version of the code has expanded capabilities for performing uncertainty analysis but little is covered in the documentation or was commented upon in the three presentations.

Resolution:

Main Report Section 5.0, "Offsite Consequence Analyses," has a short discussion of the modeling improvements and related prominent features and expanded capabilities used in SOARCA, and additional references to Main Report Section 5.0 will be added to the Appendices.

70. Page 7

Comment:

Reporting of additional consequence measures – In addition to the conditional and absolute early/latent health effect risks reported in the SOARCA study, other metrics would be advised. The uncertainty quantification and sensitivity analysis presentation in March 2010 indicated that land contamination was being considered, and would be very useful. It is advised to clarify whether forced deposition is used, and if so, the inner and outer radii the feature is employed. To compare to earlier studies, the metric of population dose over the fifty-mile region would be a useful consequence measure to complement land contamination. This is appropriate because deposition with distance would trend inversely with inhalation doses.

Related Comment (in December 3 memo, p. 10)**:**
Reporting of additional consequence measures – In addition to the conditional and absolute early/latent health effect risks reported in drafts of the SOARCA study, other metrics are advised. To be able to compare to earlier studies, the metric of population dose over the fifty-mile region would be a useful consequence measure to complement land contamination. The uncertainty quantification and sensitivity analysis presentation in March 2010 indicated that land contamination was being considered, but nothing to my knowledge was covered on this topic in the October meeting. The lower deposition velocities used in SOARCA compared to past studies, have implications for both populations dose and the land contamination consequences.

The reduced and delayed source terms compared to SST1 from the Sandia Siting Study, in the cases considered from SOARCA, provide Level 2 insights. Examples of these comparisons are shown in Tables 21 and 32 in the plant-specific reports for Peach Bottom and Surry, respectively. In addition, the LCF risk comparison of SST1 to the Peach Bottom and Surry, and other SST1 results (Peach Bottom: Figures 89 and 90; Tables 30-32) (Surry: Figures 181 and 182; Tables 46-48) accidents are useful for bringing together improvements to the health effects models, as well as smaller source terms and better modeling of countermeasures. Adding land contamination and populations dose metrics out to the 50-mile radius would produce a more immediate measure of improvements to our consequence understanding and bridge the gap from the Level 2 to the latter parts of the Level 3 insights.

Resolution:
There is no forced deposition used within the modeled 50-mile region. Historically, deposition velocities were overestimated. Current values are recommended based on the past separate MACCS peer review.

Additionally, in Appendix A, a comparison to SST1 was added to the report (as noted in the resolution to comment 63 above).

The SOARCA team recognizes that other metrics may have value, however, the scope of SOARCA has been focused on those metrics which we believe to be most important (e.g., public health effects expressed as individual risk, which is the metric that the NRC uses for its quantitative health objectives [QHOs]).

December 3, 2010 Peer Review Memorandum

71. Page 11

Comment:
Uncertainty analysis study – There are many major improvements that the SOARCA study has discussed during the Peer Review meetings and have been documented in the NUREG drafts. However, I fear that discussion of these improvements will be met by a high level of technical questioning that can only be satisfied with a comprehensive uncertainty analysis. This is a major drawback in the current study's body of information and ought to be completed before the technical reports are finalized.

Resolution:
The uncertainty analysis is underway and draft results are expected to be available in mid 2012. The numerous sensitivity analyses that have been completed, of which many results will be included in the SOARCA Main Report and Appendices, should provide confidence in the best estimate results.

72. Page 12

Comment:
Presentation and labeling in the SOARCA Documents - To improve the likelihood that the public will interpret the SOARCA study as intended, several recommendations follow:
(1) Aim for uniformity and consistency in the labeling of conditional and absolute risk figures and tables. I suggest conditional risk per event (LCF/event) and for absolute risk (LCF/reactor-year). Currently, both types of risk are labeled the same in figures and tables.
This situation may lead to incorrect interpretations by the reader. (2) Label LCF bar charts with Acute phase and Long-term phase rather than EARLY and CHRONC. The analysis is using MACCS2 as a tool to evaluate the relative importance of the short-term and long-term phases, and it should be made transparent that this is the case. Use of MACCS2 terminology in the results gives the appearance that the results are more characteristic of the manner in which the code was run, and not reflective of the post-release phases. (3) Select two of the four health

effect (dose truncation) models rather than present results from all four models. The LNT model appears to be bounding in all cases. It is recommended that this model be retained along with the one that tends to predict the most non-conservative health effect risks of the three alternative dose truncation models, i.e.,
- Health Physics Society recommendation (5 rem/year and 10 rem lifetime)
- ICRP Report 104 (10 mrem/year)
- U.S. Average Background (620 mrem/year).

Labels to figures and tables should reflect the dose truncation models with a short-hand notation of “LNT”, “HPS”, “ICRP 104”, and “U.S. Bkg. Ave.”.

Resolution:

The SOARCA team plans to modify the presentation of results as recommended, with the exception that we will retain 3 of the dose models. The 10 mrem dose truncation results are similar to the LNT and are also always slightly less than the LNT results which is why they are not included. A note explaining this will be added to the report. The model that gives the lowest latent cancer fatality risk varies (Health Physics Society recommendation or U.S. Average Background).

JOHN STEVENSON

April 30, 2010 SOARCA Peer Review Draft Report

73. Page 38

Comment:

A basic concern in his evaluation is the potential for liquifaction of soil or other foundation failure associated with seismic induced cyclic motion resulting in large vertical differential displacement of the containment or adjacent structures resulting in rupture or significant leakage of one or more of containment penetrations. A secondary consideration, and with much less probability, is the development of a fissure in the foundation media under the containment or adjacent structure propagating to the surface below the containment or other power plant structures resulting in their foundation failure. The potential for liquification induced failure is limited to saturated cohesionless soils while potential fissure failures are not so limited. In addition to containment penetration failure, fissure type failure if credible might cause simultaneous failure of the containment basemat and supports of the reactor coolant system SSC. It is understood that the types of foundation failure just described resulting from earthquakes at the median 10^{-5}/yr or mean 10^{-4}/yr probability of exceedence level have been negated by design measures such as use of engineered backfill hence; were not, nor should they have been, considered in design.

Related Comment (p. 39)**:**

Given that the site would liquefy, it would be necessary to evaluate the effect of such liquification on the leak tightness of the containment.

Related Comment (p. 40; repeated in part in December 3, 2010 memo)**:**

There may also be other NPP sites where liquification and associated differential displacement, which could cause containment penetration failure which could provide a significant containment leak path, cannot be ruled out at the 10^{-6} or 10^{-7}/yr seismic probability of exceedence levels.

As a result of the potential for liquefaction at the Surry site, it is my recommendation that a follow up on the SOARCA study be conducted which considers seismic induced soil liquefaction, consolidation and possible foundation failure which could lead to early containment be conducted. The primary concern associated with liquefaction or consolidation is that differential settlements of the containment or adjacent buildings may exceed the capacity of even a single penetration to resist significant leakage of the typically more than 100 such penetrations in the containment which could lead to early containment bypass.

Resolution:

NRC is considering future research on soil liquefaction and its effects on nuclear plant structures. The potential effect of increased leakage during the technical specification (TS) leakage phase was considered and it was judged that the effects of the later containment failure overwhelm the effects of any increased leakage during the earlier TS leakage phase. A sensitivity study on increased leakage is included in Appendix A.

74. Page 40

Comment:

A second concern raised by Dr. Stevenson was the potential for hydrogen detonation resulting from DBA hydrogen generation rather than a hydrogen deflagration. Hydrogen detonation studies have indicated a dynamic pressure buildup of 2 to 3 times the containment static design pressure which is at or very near the static failure pressure for the containment. Given the dynamic nature of the detonation this might result in a dynamic load factor depending on duration of the pressure load relative to the period of the containment structure which would amplify the peak pressure loads.

Related Comment (December 3, 2010 Memo, p. 13)**:**

I got the impression at the meeting that a containment hydrogen detonation as compared to a deflagration as the result of a LOCA that could lead to early containment failure was not an impossibility; hence, would be an issue at the very low probabilities of exceedence associated with SOARCA.

Resolution:

Studies were conducted on hydrogen detonation and deflagration and are included in the report (e.g., Section 5.2.3, “Uncertainties in the Hydrogen Combustion in the Mitigated STSBO” in Appendix B). According to MELCOR analysis, hydrogen detonation could produce substantial pressure loads, however the sprays are effective at settling airborne aerosols before detonable quantities could be formed that could fail the containment. The resulting fission product release would consist of only noble gases and would not be expected to substantially increase the offsite health consequences relative to the base case calculation. Accumulations of hydrogen (>15%) have the potential to produce detonations that, as Dr. Stevenson points out, produce dynamic loads which can exceed the capacity of the containment structure. The consequences of such events are dependent on the quantity of airborne fission products at the time. The Surry sensitivity studies suggested that airborne radioactivity at the time of detonations were in fact low (see Appendix B Section 5.2.3). The SOARCA team will discuss this at the next planned Peer Review Committee meeting.

KAREN VIEROW

April 30, 2010 SOARCA Peer Review Draft Report

75. Page 72

Comment (repeated in December 3, 2010 Memo)**:**

The primary objective of the SOARCA project is stated in several locations of the SOARCA document and in presentations to the Peer Reviewer Committee to be a "best estimate evaluation of the likely consequences of important severe accident events …" The first such claim appears in the Abstract of the Summary Report. Other locations such as the Abstract of Appendix B state that "This study has focused on providing a realistic evaluation of accident progression, source term and offsite consequences…"

It is suggested that the current evaluations are not entirely best-estimate and that care be taken in the SOARCA documents to qualify this claim. A claim to *more best-estimate*, or *more realistic,* results than produced by earlier analyses is appropriate.

While the SOARCA team has done a commendable job of enabling more realistic evaluations of severe accident consequences, several conservatisms have, in fact, been retained. Many of these conservatisms are, in the judgment of this peer reviewer, reasonable and should be discussed collectively in a visible location within the SOARCA document.

One example of conservatism is the assumption for Surry that 8 hours would be required to transport a portable diesel-driven pump and connect it to plant piping following a large seismic event (Appendix B, Section 3.1.3 Mitigative Actions). The licensee staff estimates that 2 hours would be required. A first reading may leave one with the impression that excessive conservatism has been invoked. Upon study of the event timing for mitigated events, one sees that the event sequence does not extend to the containment until 7 hours 16 minutes for the mitigated short-term blackout or 7 hours 30 minutes for the mitigated short-term station blackout with thermally-induced steam generator tube rupture. Containment spray is initiated at 8 hours for these two scenarios and sprays are not needed for the other Surry scenarios. Earlier spray activation should have some effect upon the severe accident progression, with respect to containment pressurization and retention of fission products. Discussion of the conservatism would be useful.

Resolution:

The SOARCA team understands the concern related to the terms “best-estimate” and “more realistic” results. The SOARCA team, however, did strive to seek the “best-estimate” results. See comment resolution 76 for additional discussion.

76. Page 73

Comment (repeated in December 3, 2010 Memo)**:**

Because a “best estimate evaluation” is a stated primary goal of the SOARCA project, this reviewer suggests that a compendium of conservatisms be included within the SOARCA documentation, perhaps as an appendix or within a discussion section on the extent to which SOARCA objectives have been met. Within this appendix or discussion, the argument should be made that inclusion of some conservatism is warranted. Two reasons for justification come to mind. Firstly, conservatism is one method for treating uncertainties. Secondly, if a nonconservative approach were to be taken, the SOARCA project could be interpreted by the public as being overly optimistic about nuclear safety and thereby lose credibility.

Another suggestion is to perform a calculation in which the conservatisms are removed. For example, have the containment sprays operable from two hours and observe the differences in results. This approach is analogous to performing the consequence analysis using actual weather condition from a typical day, instead of specifying conservative or time-averaged conditions.

In summary, care should be taken in public documents and presentations to qualify the degree to which the analysis methods and results can be regarded “best-estimate” or “realistic”. The qualified claim of *more realistic* evaluations seems appropriate

Resolution:

A guiding principal of the SOARCA analysis has been to avoid conservatisms and make every attempt to provide best estimate results. Nevertheless, there are conservatisms that are still reflected in the SOARCA results. For example, the assumption of mid-day population motion during a weekday to present the most challenging evacuation scenario. At the same time emergency response organizations are assumed to be staffed at nighttime levels.

Conservatisms were included in the analyses where it was judged that the additional work necessary to justify a better estimate was not worthwhile, most notably because it was not expected to materially affect the results. While we do not plan to include a compendium of conservatisms in the SOARCA documentation, conservatisms are apparent in the SOARCA reports. For example, Section 6 of appendices A and B discuss the assumption of mid-day population motion during a weekday.

December 3, 2010 Peer Review Memorandum

77. Page 14

Comment:
Referring to the second objective, consideration of power uprates and higher core burnup in the MELCOR analysis is unclear.

Resolution:
The MELCOR decks reflect the current status of the Peach Bottom and Surry plants, and thus reflect the approved power uprates and higher core burnups at those plants (compared to earlier assessments). For the purposes of estimating the available radionuclide inventory, SOARCA analyses assume the accident occurs mid-way through a typical operating cycle. SOARCA analyses used actual core design and performance data for three consecutive contemporary cycles to build the MELCOR and ORIGEN input data. The approach is discussed in the main report Section 4.3.4, “Radionuclide Inventory,” in Appendix A Section 4.7, “Radionuclide Inventories and Decay Heat,” and Appendix B Section 4.5.2, “Initial Radionuclide Inventory.”

78. Page 14

Comment:
Referring to the fourth objective and communication with the general public, a document written in layman’s terms is needed

The SOARCA team has a separate explicit task from the Commission to ensure that SOARCA is communicated effectively to the public. A plain-language brochure will be developed for release at the same time as the SOARCA reports.

JACQUELYN YANCH

Comment report received May/June 2010

79. Page 1

Comment:
We know very little about the health impact of low dose and, more particularly, of low dose-rate radiation; we should make every effort to redress this lack of understanding so that the public can be appropriately guided as they deal with the aftermath of a severe reactor accident.

Related Comment (p. 16)**:**
As mentioned in the SOARCA documentation, little guidance as to how to estimate the likely health impact of low dose, low-dose-rate radiation exposure is provided by the national and international committees who examine available data, and the approach we borrow from radiation protection risk estimates is *not appropriate* for use in dealing with long-term exposures due to radionuclides in the environment. Therefore, while it is not the role of the NRC to dictate how the RHDLs should be set, the NRC and the industry should take a stronger position on determining the true nature of radiation-related health effects at the dose-rates anticipated following a severe nuclear power plant accident. It should be a priority.

Resolution:
The SOARCA team agrees that the state-of-the-art in low dose assessment can be improved; however the team feels that the SOARCA study is consistent with the current state-of-the-art available, and represents an improvement over earlier estimates.

80. Page 2

Comment:
What is the health impact of the return-home dose-rates? We don't know yet. None of the data we use in estimating radiation-induced health effects were obtained at the doses and dose rates similar to those encountered upon returning home. Therefore we have essentially *no* understanding of the potential health consequences of these radiation conditions. More importantly, we have no understanding of the health impact of the radiation dose-rates that were avoided by staying away from home for so long.

Resolution:

In the SOARCA study, the SOARCA team addressed the uncertainty of health effects induced by low doses and dose rates by evaluating a set of dose-truncation values. We believe that at least to some small degree, we have thereby shed some light on this issue and the extent to which it affects risk predictions. In addition, we plan to explore the effect of the return criteria in the uncertainty analysis for the project.

81. Page 16

Comment:

Some countries operate a registry for tracking annual occupational dose for all radiation workers [35]. The existence of such a registry makes it feasible, at some time in the future, to examine health effects as a function of doses received. In most case, however, occupational doses are much *smaller* than individual medical doses [14,29] and therefore any health impact of occupational radiation may never be discernable from the potential effects of the larger medical doses. It makes more sense, therefore, to record our medical doses and to store these in a database. This we do not do.

A couple of decades ago the doses received from diagnostic radiology were relatively small and experienced by relatively few individuals. Today, however, radiological exams are used for addressing a much broader range of medical questions and are performed on a much bigger fraction of the population. More important is the fact that we've begun to make routine use of the more dose-intensive procedures of x-ray computed tomography (CT) and interventional fluoroscopy [17]. The result is that the average US resident receives as much radiation dose from diagnostic radiology procedures as from all natural background radiation sources, combined. Thus, on a routine basis, and for a variety of reasons, we deliberately and carefully irradiate most members of the U.S. population, exposing them to a wide range of doses depending on the reason for the exam, the part of the body being imaged, and the patient's body thickness. It makes sense to maintain a registry of radiation doses for everyone irradiated. This registry would not be a "de-identified" patient radiation dose data-base (as proposed recently by the FDA as a starting point for establishing consistent exam parameters across medical institutions [36]), but a registry that allows tracking of an individual's dose over time and, ultimately, for correlation of dose with disease or health status many years later.

Resolution:

The existence of a database or multiple databases that track occupational and medical doses would further the knowledge on health effects as a function of dose received over protracted periods. The SOARCA team agrees that the state-of-the-art in low dose assessment can be improved; however the team feels that the SOARCA study is consistent with the current state-of-the-art available, and represents an improvement over earlier estimates.

82. Page 18

Comment:

For understanding the impact of chronic, low dose-rate radiation we can examine the many regions of the world whose inhabitants are already living with the dose rates represented by the RHDLs (and higher), and have been for many generations. Residents of high background radiation areas (HBRAs) do not appear to suffer adverse effects from these dose-rates (and in some cases appear to be healthier and living longer than those living in nearby control areas with lower radiation levels [15,37,38]). Such comparisons, however, often suffer from small sample size, incomplete dosimetry, and a lack of uniformity between studies that prohibit combining of the data. Many comparison studies are ecologic in design in which dosimetry data are aggregated over the entire populations; this type of study is subject to several bias and correlation problems which do not arise in cohort or case/control studies where information for each individual subject (dosimetry, confounding factors, etc.) is available [39]. UNSCEAR has suggested that only cohort or case/control studies are suitable for quantification of radiation risk [40].

Resolution:

Dr. Yanch in her comments has identified a number of known limitations in the current dose response models used to predict radiogenic human health risks. Many of these issues arise from the extrapolation of health risks of populations exposed to high-dose, dose-rate radiation—e.g., Japanese atomic bomb survivors— to populations exposed to low-dose and low dose-rate radiation. Future consequence analyses could benefit from research along the lines suggested by Dr. Yanch.