PEER REVIEW OF THE STATE-OF-THE-ART REACTOR CONSEQUENCE ANALYSIS (SOARCA) PROJECT

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Prepared by:
Ken Canavan, Electric Power Research Institute (EPRI);
Bernard Clément, Institut de Radioprotection et de Sûreté Nucléaire;
Jeff R. Gabor, ERIN Engineering;
Robert E. Henry, Fauske and Associates;
Roger B. Kowieski, Natural and Technological Hazards Management Consulting, Inc. (NTHMC);
David E. W. Leaver, WorleyParsons Polestar;
Bruce B. Mrowca, Information Systems Laboratories;
Kevin R. O'Kula, URS Safety Management Solutions;
John D. Stevenson, JD Stevenson Consulting Engineer Company;
Karen Vierow, Texas A&M University;
Jacquelyn C. Yanch, Massachusetts Institute of Technology

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1. Introduction

1.1 Importance of SOARCA Peer Review

The Nuclear Regulatory Commission (NRC) is conducting the State-of-the-Art Reactor Consequence Analysis (SOARCA) to update evaluations of hypothetical severe accident progression and offsite consequences in nuclear reactors. SOARCA originated in efforts to assess nuclear power plant response to security-related events. The project aims to provide more realistic assessments of the risks posed by nuclear power plants by reducing excessive conservatisms in earlier evaluations and incorporating the most recent plant information and analysis technologies. An anticipated result is a major change in the general public's perceptions of nuclear reactor safety.

In this context, the SOARCA incorporates insights and analysis techniques that are significantly different from those used in previous consequence analyses, along with updated information on plant improvements and security-related enhancements. The advances and changes in these areas represent major improvements in the knowledge of severe accidents and risks to the public health.

The SOARCA Peer Review Committee was appointed to provide an independent review of these updated analyses. Technical experts from industry, consulting, academia, and research laboratories have been assembled to assess all aspects of the project in an impartial manner and provide guidance and suggestions. The Committee possesses extensive knowledge regarding plant design, operation and maintenance, safety and security-related equipment, severe accident phenomenology, emergency preparedness, and radiological health consequences and analysis thereof.

The SOARCA integration of analysis tools and techniques, along with incorporation of recent plant improvements and security-related enhancements, represents a new application of the state-of-the-art analysis techniques. The Peer Review Committee fills the essential role of reviewing the technical work performed under the SOARCA. The scope of review includes correctness of information used, assumptions, analysis methodologies, application of current standards, and practices and interpretation of results.

1.2 Peer Review Objectives

The main objective of the Peer Review Committee is to provide independent reviews by each Committee member of the technical work conducted within the SOARCA project. The primary focus is to assure that the SOARCA study is technically accurate.

Guidance with respect to specific issues, as requested by NRC staff, and comments on the effectiveness of presentation within the SOARCA NUREG documents to the public have also been offered by the Committee members.

1.3 Peer Review Committee Members

The Peer Review Committee is comprised of the following eleven technical and scientific experts.

- Ken Canavan, a Senior Program Manager in the Risk and Safety Management (RSM) program for the Electric Power Research Institute (EPRI), reviewed accident sequence selection and progression. For the last 24, years he has worked in the risk and safety discipline for nuclear utilities, consultants, and most recently EPRI on the development of probabilistic risk assessments (PRA), PRA methods, risk-informed applications, peer certification process, and several unique applications of risk technology. Mr. Canavan earned a Bachelor of Engineering in Chemical Engineering with a nuclear sequence from Manhattan College.
- Bernard Clément, senior expert at France's Institut de Radioprotection et de Sûreté Nucléaire, reviewed accident progression and radiological release. His 30-plus years in nuclear safety research have examined light-water reactor design-basis and beyond design-basis accidents as well as liquid-metal fast-breeder reactor safety. Clément has chaired the scientific analysis working groups of the Phebus FP and International Source Term Programs. He is a graduate of the French Ecole Centrale de Paris.
- Jeff R. Gabor, vice president of the risk management group for ERIN Engineering, reviewed accident progression and radiological release. In more than 25 years of nuclear power plant safety experience, he has worked on numerous Level 2 Probabilistic Safety Analysis (PSA) updates, supported several utilities' severe accident and thermal-hydraulic analyses, and developed severe accident mitigation guidance, and he was a principal author of the Boiling Water Reactor Modular Accident Analysis Program. He earned a Bachelor of Science in nuclear engineering and a Master of Science in mechanical engineering from the University of Cincinnati.
- Robert E. Henry, senior vice president and co-founder of Fauske and Associates, reviewed accident progression and radiological release. Henry's more than 40 years of nuclear safety and engineering experience include work on light-water reactor response to severe accidents and severe accident management guidelines for all commercial U.S. reactors. He earned his bachelor's, master's and doctoral degrees in mechanical engineering from the University of Notre Dame.
- Roger B. Kowieski, president of Natural and Technological Hazards Management Consulting, Inc. (NTHMC), reviewed off-site emergency planning and response. His 30 years of experience cover a very broad spectrum of emergency planning and preparedness including reviews of radiological and chemical hazards assessment reports; development of protective actions decision making trees; development of lesson plans and trainee manuals; conducting of training sessions for facility personnel; design and evaluation of Radiological Emergency Preparedness (REP) exercises for nuclear power plants for FEMA. While with FEMA until 1988, he served as a FEMA expert witness before the NRC Atomic Safety and Licensing Boards (ASLBs) in connection with licensing actions on the Indian Point and Shoreham Nuclear Power Stations. He currently serves as the Regional Coordinator, assisting FEMA Region 3 in the planning

and execution of all REP exercises in this region. Kowieski earned his Master of Science degree in Environmental Engineering from Wroclaw Polytechnic, Wroclaw, Poland.

- David E. W. Leaver, a senior vice president and principal at WorleyParsons Polestar, reviewed radiological release, emergency response, and offsite radiological consequences. He performed some of the earliest PRA studies of nuclear plants during his more than 30 years in reactor safety, risk assessment, radiological source term and accident analysis, emergency planning support to the nuclear industry, and meteorological analysis. Leaver earned his Bachelor of Science in electrical engineering from the University of Washington, and earned his Master of Science in engineering economic systems and a doctorate in mechanical engineering from Stanford University.
- Bruce B. Mrowca, vice president and manager for nuclear system analysis operations of Information Systems Laboratories, reviewed probabilistic risk assessment (PRA) sequence selection and mitigation measures. His more than 25 years of experience in commercial nuclear power include PRA development and application, instrumentation and control design and fire protection analysis. He earned his Bachelor of Science in electrical engineering from the University of Maryland.
- Kevin R. O'Kula, of URS Safety Management Solutions, reviewed offsite radiological consequences. For more than 26 years O'Kula has examined topics including accident and consequence analysis, source term evaluation, commercial and production reactor PRA and severe accident analysis, and safety software quality assurance. He earned his Bachelor of Science in applied and engineering physics from Cornell University, and his Master of Science and doctorate in nuclear engineering from the University of Wisconsin.
- John D. Stevenson, a senior consultant at JD Stevenson Consulting Engineering Company, reviewed structural and seismic issues. His 35 years of experience include developing structural and mechanical construction and design criteria for qualifying nuclear power plants, structures, systems and components applications to resist extreme natural and man-induced hazards. Stevenson earned his Bachelor of Science in civil engineering from Virginia Military Institute, and his Master of Science and doctoral degrees in civil engineering from Case Institute of Technology. He currently is chairman of the Technical Advisory Committee to the International Atomic Energy Agency Seismic Safety Center.
- Karen Vierow, associate professor of nuclear engineering at Texas A&M University, chaired the Committee and reviewed severe accident modeling. Her 20 years of experience in nuclear engineering focus primarily on thermal hydraulics, reactor safety, severe accidents and reactor design. Vierow earned a Bachelor of Science in nuclear engineering from Purdue University and a Master of Science in nuclear engineering from the University of California at Berkeley. She earned her doctorate in quantum engineering and system sciences from the University of Tokyo.
- Jacquelyn C. Yanch is Professor of Nuclear Science and Engineering at the Massachusetts Institute of Technology where she has been a member of the faculty since 1989. Yanch reviewed the off-site radiological consequences. Her research deals with the production, detection, applications, and health effects of ionizing radiation and

involves both physical experimentation and computational dosimetry applied to human irradiations. Current experimental work involves long-term irradiations of cell and animals at low dose-rates. In 2009 Professor Yanch also became a member of the MIT Department of Biological Engineering. Yanch has served on the MIT Reactor Safeguards Committee and the Committee on Radiation Exposure of Human Subjects and has been a member of the MIT Radiation Protection Committee for 20 years.

1.4 Report Organization

Section 2 of this report describes the Peer Review Committee charter and scope of review. The coverage of SOARCA topics is explained and the peer review approach is discussed.

Each Committee member's individual assessment of the SOARCA effort is included in Section 3.

The Appendices include comments and suggestions that the Peer Review Committee members have provided to the SOARCA point of contact throughout the review process.

2. Peer Review Process

2.1 Committee Charter

The Peer Review Committee's charter is to provide independent reviews of the technical work conducted by the NRC and Sandia National Laboratories for the SOARCA project. The primary focus is to assure that the SOARCA study is technically accurate. The Committee is also to assess whether the conclusions and the Executive Summary are supported by the underlying technical work presented in the draft SOARCA NUREG report.

Guidance with respect to presentation within the SOARCA NUREG documents of the results to the general public may also be offered by the Committee.

The final deliverable is this technical report documenting the findings of individual Committee members.

The Committee began its work in July 2009 and submitted the final version of this report in January 2012.

2.2 Peer Review Scope

The scientific and technical experts on the Committee were requested to assess the methodological approach, underlying assumptions, results and conclusions obtained for Peach Bottom and Surry reactors. The Committee members also commented on the presentation of the SOARCA evaluations within the SOARCA NUREG documents.

The documents reviewed included draft SOARCA NUREG documents, presentation materials provided at Peer Review Committee meetings, comment resolution documents and supporting documents that were supplied at the Committee's request. The draft SOARCA NUREG document dated Dec. 23, 2011 (Main Report), Oct. 12, 2011 (Volume 1) and Nov. 17, 2011 (Volume 2) is the latest version available to the Committee at the time of preparation of this report.

The scope of the review does not include the Uncertainty Quantification and Sensitivity Analysis. However a proposed Uncertainty Analysis methodology was presented to the Committee on Oct. 26-27, 2010 and a peer review guidance memo was requested to be included in this report as an attachment. The parameters and their distributions to be used in the Uncertainty Analysis were presented to the Committee on January 5, 2012.

The current effort also does not include editorial review of the SOARCA documents.

2.3 Coverage of SOARCA Topics by Committee Members' Areas of Expertise

Peer Review Committee members reviewed the SOARCA according to their areas of expertise as follows:

Accident sequence selection
Ken Canavan
Bruce Mrowca

Accident progression
Ken Canavan
Bernard Clément
Jeff Gabor
Robert Henry

Mitigation measures
Jeff Gabor
Robert Henry
Bruce Mrowca

Radiological release
Bernard Clément
Jeff Gabor
Robert Henry
David Leaver

Off-site emergency planning and response
Roger Kowieski
David Leaver

Off-site radiological consequences
David Leaver
Kevin O'Kula
Jacquelyn Yanch

Seismic issues
John Stevenson

Structural issues
John Stevenson

Probabilistic Risk Assessment applications
Ken Canavan
Bruce Mrowca

Severe accident modeling
Jeff Gabor
Robert Henry
Karen Vierow

2.4 Peer Review Approach and Methodology

Five meetings were conducted between the Peer Review Committee members and the SOARCA team. Prior to each meeting, SOARCA documentation was transmitted to the Committee for review.

The first meeting between the Committee members and the SOARCA team was held in Rockville, MD on July 28-29, 2009. A draft of the SOARCA NUREG document, dated July 2009, was received for review prior to the meeting. The SOARCA team presented the project to the Committee members and initial comments and questions were discussed verbally. Following the meeting, the Committee provided written comments on the SOARCA document and information presented at the two-day meeting, as documented in Appendix A.

The second meeting was conducted on September 15-16, 2009 in Bethesda, MD. Prior to this meeting, supplemental materials including reports of MELCOR and MACCS external review committees, the 1982 Sandia Siting Study and a memo from Dana Powers on fission product retention in steam generator tubes were transmitted to the Committee members. The SOARCA team presented the latest results to the Committee members and comments and questions were discussed verbally. Following the meeting, the Committee provided written comments on the SOARCA document and information presented at the two-day meeting, as documented in Appendix B.

The third meeting was conducted on March 2-3, 2010 in Rockville, MD. A draft of the SOARCA NUREG document, dated February 14, 2010, was received for review prior to the meeting. Presentations by the SOARCA team on the first day focused on comment resolution and plans for Uncertainty Quantification and Sensitivity Analysis. Through discussion with the SOARCA team, the latter effort was determined to be outside of the Committee's charter. The second day of meetings was primarily for discussions amongst the peer reviewers and small-group meetings with members of the SOARCA team, as requested by the peer reviewers.

Several action items arose from this meeting. First, the Committee members were asked to provide written comments on the description of the SOARCA in the draft NUREG. These comments are included in Appendix C. Second, issues arose for which the SOARCA team requested guidance on a time scale shorter than that for preparation of the Committee's final report. This memo is attached as Appendix D. Third, the Committee members were asked for their insights into the Uncertainty Quantification and Sensitivity Analysis, an issue which several members were interested in but which was determined to be outside of the review scope. This memo is attached as Appendix E.

The fourth was held on October 26-27, 2010 in Rockville, MD. For meeting preparation, the reviewers received a draft plan for the Uncertainty Analysis, dated Oct. 19, 2010, a draft NUREG/CR on uncertain input parameters for use in off-site consequence analysis codes and an ORNL report documenting an uncertainties in cancer risk coefficients. The SOARCA team discussed the June 2010 ACRS meeting, the proposed Uncertainty Analysis technical approach and resolution of peer review comments from earlier meetings. On the second day, the peer reviewers discussed completion of this report.

Action items arising in this meeting were to: prepare a list of unresolved review comments; draft a guidance memo on the Uncertainty Analysis; and finalize this report. The guidance memo is included herein as Attachment F.

The fifth and final meeting was held in Rockville, MD on Dec. 6-8, 2011. Prior to this meeting, the reviewers received the draft SOARCA NUREG document dated Nov. 29, 2011 (Main Report), Oct. 12, 2011 (Volume 1) and Nov. 17, 2011 (Volume 2). The SOARCA team presented changes in SOARCA analyses and results since the previous meeting on the first two days. On the third day, a description of the MELCOR code validation effort was provided and discussed.

The SOARCA team provided a written response to reviewer comments to date on July 29, 2011. A teleconference between the SOARCA team and the peer reviewers was conducted on Sept. 5, 2011 to discuss the responses and clarify whether the reviewers had any remaining issues.

Another teleconference was held on Jan. 5, 2012 to explain to the peer reviewers the selection of parameters being used in the Uncertainty Analysis and their distributions. The reviewers have not seen the results or conclusions of the Uncertainty Analysis at the time of preparation of this final report.

The final deliverable of the Peer Review Committee is a report to the SOARCA team documenting the technical findings of the individual peer reviewers. The report has been assembled and coordinated through the Peer Reviewer Committee chair.

A consensus opinion of the Committee has not been pursued or documented during the review process. All of the written materials described above, which were provided to the SOARCA team by the reviewers, have been assembled by and coordinated through the Peer Review Committee chair. Each reviewer's assessment of SOARCA has been transmitted as received, without editing or other modification.

3. Individual Assessments from Peer Review Committee Members

Individual assessments of the SOARCA by each Peer Review Committee member are included in the next page, in alphabetical order by reviewers' last names. These assessments are included exactly as they were transmitted to the Chair of the Committee and have not been edited in any manner.

Individual Input from Review of State-of-the Art Consequence Analysis (SOARCA)

Ken Canavan, Senior Program Manager
Risk and Safety Management (RSM)
Electric Power Research Institute

Overview

As stated in "State-of-the-Art Reactor Consequence Analysis (SOARCA) Project Methods," the overall objective of SOARCA is to develop a body of knowledge regarding the realistic outcomes of severe reactor accidents. Supporting objectives include:

- incorporate plant improvements and updates not reflected in earlier assessments
- incorporate state-of-the-art integrated modeling of severe accident behavior
- evaluate the potential benefits of recent security-related mitigation improvements in preventing core damage and reducing an offsite release should one occur;
- enable the NRC to communicate severe-accident-related aspects of nuclear safety to stakeholders; and
- update quantification of offsite consequences found in earlier NRC publications such as NUREG/CR-2239, "Technical Guidance for Siting Criteria Development."

The SOARCA study has largely met its objectives. Plant improvements and significant changes in design, maintenance, and operational practices that have been implemented over the past twenty years have been incorporated and reflected in the SOARCA models. State-of-the-art severe accident modeling has not only been implemented in SOARCA, but the state-of-the-art has been extended by SOARCA through the significant amount of technical work and research developed for, and implemented in, the study. SOARCA has addressed the benefits of the improvements at the plants including the recent security related enhancements. In the area of severe accident communication, the technical community will benefit from the developments in SOARCA; the benefits and efficacy of communication with other stakeholders was beyond the scope of this review. The last objective, the quantification of offsite consequences was addressed, for the more likely accident sequences, through the selection of the most significant accident scenarios and the detailed modeling of the radiological consequences. Additional discussion of this last objective is provided in the following paragraphs.

While the goals and objectives of SOARCA appear to be largely achieved, and in some cases exceeded, there are some observations worthy of note. This reviewer's comments are limited to the assigned topical area of accident sequence analysis.

Consequence Analyses

The overall goal of SOARCA, as supported by the objectives of the study, is to develop current and realistic estimates of the potential offsite consequences from the more likely severe accidents for an operating nuclear power plant. However, as is the case of all consequence analyses, SOARCA focuses on only the most significant accident sequences for detailed consequence analysis. While effort in SOARCA is expended to attempt to ensure that that the most significant sequences, in terms of both frequency of occurrence and related consequences are chosen, there are always questions as to the rigor of this process in any analyses of this type. As such, some may question the impact of non-dominant or individually non-significant accident sequences or whether this approach is sufficient for the application chosen.

For example, there is the possibility that certain accident sequences, while not dominant, may have higher risk than would be indicated by the frequency of the accident sequences due to an increased consequence. While these sequences may not dominant the risk, in terms of either frequency and/or consequence, they could be contributors. Combinations of several lower order sequences could have higher consequence than the specific accident sequences considered by SOARCA and could contribute to higher risk. While SOARCA did indeed capture the most likely sequences and did accurately capture the consequence from these sequences this approach may not be sufficient to demonstrate completeness.

This issue of "completeness" is common in consequence analyses. That is, for consequence analyses, it can be difficult to demonstrate completeness. The benefits of a frequency-weighted approach, such as a level 3 probabilistic risk analysis (PRA), is that the accident sequence frequencies and consequences can be used in the determination of risk. The results of the PRA accident sequence frequencies and the related consequences can be evaluated both individually and collectively. However, the frequency-weighed approach is also susceptible to instances where the results can be misinterpreted, taken out of context, or manipulated without proper basis. This reviewer believes the benefits of demonstrating completeness outweigh the potential for intentional or unintentional misuse.

A level 3 PRA performed for a SOARCA plant would have the benefit of reduced resources (due to work performed for SOARCA) as well as the benefits of validation of the SOARCA approach and demonstration of completeness.

Plant-Specific Nature of SOARCA

The SOARCA analysis was developed by applying the methodology to two plants, Surry and Peach Bottom. Such a plant-specific derivation incurs both positive and negative aspects. With plant-specific information, plant-specific conclusions can be drawn based on the design features, maintenance and operation practices at that particular site. Conversely, because these plants do not encompass all of the design, maintenance and operation practices across the nuclear fleet – both those that reduce consequences and

those that might increase consequences – some conclusions are likely applicable to that site only and the results may not be typical.

For example, because the Peach Bottom drywell does not have a curb, direct containment heating via corium contact with the liner is possible. In other BWR Mark I containments, the liner may prevent or reduce the likelihood of corium contact with the liner.

While an alternative to the current approach or analysis is not recommended, it is important to emphasize that the results can be influenced in a material way by plant-specific features.

Individual Accident Sequences

As part of the SOARCA review, the accident sequences criteria used in the SOARCA study were applied generically to various accident sequences in previously published PRA studies. The conclusion of this comparison was that no new accident sequences were identified that should have been included in SOARCA. However, it should be noted that this review was informal and generic. Plant-specific application could produce different results. While the generic comparison does provide some assurance that the criteria was correctly applied, the completeness issue discussed in the "Consequence Analysis" section apply.

Safety valves and pilot operated relief valves play a significant role in the accident sequences analyzed in SOARCA. The successful operation of these valves, as well as their failure modes under beyond design basis conditions, is clearly significant in the analysis. This reviewer believes the safety valve failure modes considered in the SOARCA analysis are very likely. However, this likelihood illustrates another advantage of a frequency weighed approach where competing and important phenomena can be frequency weighted, the result is a more holistic view of risk and the key contributors.

Summary

The SOARCA analysis has met its stated primary goal of developing current and realistic estimates of the potential site-specific offsite consequences from the more likely severe accidents for operating nuclear power plant.

In addition, the other objectives of the study were also achieved including incorporation of plant improvements and updates, state-of-the-art integrated modeling of severe accidents, and incorporation of the benefits of recent security-related mitigation improvements.

However, SOARCA is a consequence study and, as such, has issues associated with demonstrating completeness. Consequence studies are limited in their ability to obtain the most utility from the final results because it is difficult to implement advancements in the technology or changes in the state of knowledge. In addition, SOARCA is plant-specific, which has the benefit of reflecting the specific plant conditions, but has the

detriment of not reflecting the range of potential designs or how these alternate designs might influence the results. In the accident sequence analysis, changes in assumptions or the state of knowledge of certain phenomena could influence the results of the analysis and further limit the usefulness of the final result.

Evaluation by B. Clément

Summary

The reviewer looked at all the documentation provided by the SOARCA project. His evaluation mainly focussed on the domains related to his personal background: (i) objectives and approach, (ii) accident scenario analysis, and (iii) uncertainty analysis. Finally, recommendations for possible work continuation are given.

The SOARCA project succeeded in achieving the objective of updating quantification of offsite consequences. This was done by using best-estimate simulation tools on a limited number of accident sequences. The selected scenarios result in containment failure, very large leakage or bypass representing a class of accidents with quite large but not early releases. This is considered as being correct, and overall, the SOARCA methodology proved to be useful.

The accident progression is calculated using the MELCOR state-of-the-art code. In the calculations, 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. The reviewer considers that uncertainties exist concerning the first failure location. This was addressed for SGTR but not for RPV failure. A recommendation in that sense was made during the review meetings. The MELCOR code does not yet incorporate all the outcomes of recent R&D on fission products behavior, especially as far as iodine is concerned. To overcome this difficulty, a superimposition of gaseous iodine source term directly coming from Phebus experimental results was superimposed to the one calculated by MELCOR. This gives consistent results for the sequences that were studied, but it might not be the case for other sequences.

Addressing the uncertainties issue within the frame of the SOARCA project will certainly increase the robustness of the results and the confidence we can have in the conclusions. Given the important amount of work needed, the project proposes to conduct the uncertainty study on one sequence for one power plant. This is considered as being acceptable and a good starting point. Besides, the methodology for uncertainty analysis is valid.

For future consideration, it is recommended: (i) to proceed to a revision of part of the SOARCA documentation according to new PRA results if their outcomes make it useful, (ii) to address other pilot plants representative of other designs using the SOARCA methodology, and (iii) to benchmark SOARCA evaluations of some selected sequences with a new MELCOR version incorporating significant new features when it becomes available.

Introduction

Given his background, the reviewer mostly focused on general documents describing the SOARCA objectives and methodology as well as on accident progression and source-term analyses. For the same reason, more input will be found for the Surry PWR than for the Peach Bottom BWR.

SOARCA Objectives and Approach

Among the different objectives assigned to the SOARCA project, the most important in the reviewer's opinion is to "update quantification of offsite consequences found in earlier NRC publications." Indeed, the quantifications in NUREG/CR-2239 were likely overly pessimistic.

The SOARCA study takes into account significant plant improvements and updates not reflected in earlier assessments and evaluates the potential benefits of mitigation improvements. In that sense, it is up to date.

SOARCA uses an integrated approach based on the use of two best-estimate simulation tools, MELCOR and MACCS2. These two codes incorporate to a large extent the current status of knowledge on severe accidents.

For fully answering the question, "Is SOARCA a best-estimate study," one needs to consider the accident-scenario selection procedure discussed in the next section.

Overall, the reviewer considers that the SOARCA approach is useful and valid.

Accident Scenario Selection

Because SOARCA is not a full Level 3 PRA study, only a limited number of scenarios has been selected. The accident scenario selection is based on Core Damage Frequency criteria. Though radionuclide release frequency criteria would have been preferable, the results of Level 2 and Level 3 PRA results made available to the project at its initiation were probably not enough numerous and/or complete to do so. As a result of the chosen screening criteria, sequences with Large Early Release Frequency were not considered due to their very low occurrence probability. All the unmitigated SOARCA scenarios result in containment failure, very large leakage, or bypass representing a class of scenarios with quite large but not early releases. Release is much smaller for mitigated scenarios. It is considered that the screening method used leads to a correct selection of scenarios.

Accident Progression and Source-Term Analysis

The accident progression is calculated with MELCOR, which is undoubtedly a state-of-the-art tool for core degradation but does not yet incorporate all the recent outcomes of research on source term.

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, which delays the progression of the accident. Secondly, this avoids any high pressure melt ejection. In addition to this base case, scenarios

with thermally induced SGTR 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.

The analysis shows that hydrogen combustion by jet ignition becomes possible after the hot leg rupture. Bounding cases are given for AICC and detonation. It would be interesting to see whether we are far from the σ criterion for flame acceleration and the λ criterion for detonation in order to evaluate them.

Again, for the Surry analysis, the releases are due to containment's overpressure. The basement failure and the associated release path were not considered. In most of the analyzed sequences, the duration between debris discharge to cavity (followed shortly by cavity dryout), and increased leakage of containment is probably sufficiently short to consider that release through the failed basement will not be an important contributor to the overall release. This might not be the case for the unmitigated long-term station blackout where this time difference is about 24 hours. This point could be addressed in the future through a sensitivity study.

As for Peach Bottom accident progression, the same general comments about MELCOR can be made. The question of uncertainties on mechanical failures is also relevant: it applies for Peach Bottom to the rupture of the main steam line.

Concerning the release of fission products from the fuel, MELCOR uses CORSOR-Booth models with diffusion coefficients adjusted on a large number of experimental data. One can consider that the results obtained are reliable. One can draw the same conclusion for the transport of aerosol in the RCS despite the fact that some phenomena are not modelled. The chemical aspects, especially for iodine, are more complex. 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 modelling lack: gaseous iodine concentrations observed in the Phebus FPT-1 experiment were added to the containment inventory. 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 than the 48 hours considered in the studies.

Uncertainty analysis

Addressing the uncertainties issue within the frame of the SOARCA project will certainly increase the robustness of the results and the confidence we can have in the conclusions. Given the large amount of work needed, the project proposes to conduct the uncertainty study on one sequence for one power plant. This is considered as being acceptable and a good starting point.

Uncertainties are generally classified in two categories: epistemic and random. In principle, their treatment should be different. However, the practical way to cope with uncertainties when using physical/numerical models is to assign a probability distribution function to a number of selected parameters and/or model options, not making any distinction between the different types of uncertainties. This is also acceptable. There is nevertheless a type of uncertainty that cannot be treated that way: it is the case when you know that some physical phenomena, potentially important, are not modelled in the tools you are using. Then a solution can be to make a sensitivity analysis by superimposing "by hand" (using side calculations and/or considerations) the hypothesized effect of such phenomena and looking at how much it

impacts the overall results of the study. An example of such an approach is what was already done for gaseous iodine using results from Phebus FP. If not giving an uncertainty, the method can allow a qualitative measurement of the impact of nonmodelled phenomena.

As for the statistical method, Monte Carlo sampling should be preferred to Latin Hypercube, not only for theoretical reasons, but also for practical ones: tools are available in MELCOR and work well.

A most important part of the work is the selection of parameters to be examined and the determination of their probability density functions. This needs to be done based on expert judgment and reviewed, not necessarily outside of the project.

At a first glance, the list of parameters presented during the March 2010 review meeting for Peach Bottom accident progression seems to be adequate. One difficulty is that some of them might not be fully independent, whereas they should be for a Monte Carlo sampling. Attention must be paid to core degradation parameters for which interdependencies are suspected by the reviewer.

Concerning the probability density functions, the choice of finite ones is supported because sampling in the tails of infinite distributions may lead to the selection of a parameter's value that falls largely outside of the validation range of the model. In addition to uniform and triangular distributions, truncated Gaussian and truncated log-normal could also be selected for some cases.

Recommendations

The objectives of the SOARCA project were not to develop a full Level 3 PRA. There is however an undeniable interest in developing Level 2 and Level 3 PRAs. Such developments, if possible, should be made in parallel with the continuation of SOARCA project. Depending on the outcomes of new PRAs, it might be useful to proceed to a revision of part of the SOARCA documentation.

The SOARCA methodology has now been applied to two pilot plants representative of two major classes of U.S. operating nuclear power plants. Before deciding on extensding it to the whole U.S. fleet, it would be interesting to address other pilot plants representative of other designs such as BWRs with Mark 2 containment or PWRs with ice-condensers containments.

The outcomes of the uncertainty analysis may have two different consequences: some aspects may appear unimportant and should be treated with fewer details in the future; on the contrary, some other aspects may appear more important than initially foreseen, and they need closer attention in the future.

Progress has been made in 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.

Update as of January 20th, 2012

This update follows the release end 2011-beginning 2012 of the latest version of SOARCA reports (main report including executive summary, appendices A and B) as well as the uncertainty analysis report, taking into account the outcomes of the December 2011 phone

conference. It also includes a short paragraph about the resolution of various comments by NRC.

Resolution of comments by NRC

The answers to comments are in general satisfactory. There is only one point still deserving attention: the likelihood of thermally-induced steam generator tube rupture for Surry. A first answer was that “it was considered incredible for the scope of the present study”. A more convincing one was that “separate multi-year work is underway , in a different NRC project, to assess he likelihood of TI-SGTR, considering updated flaw distributions and material changes”. The outcomes of the programme could be used for a possible revision, if needed, of SOARCA study.

The comments on the uncertainty studies have also been well taken into account. It is well appreciated that the possible correlations between input variables of the models will be taken into account for some of them in the Monte Carlo study. For the analysis of consequences, it is appreciated that the possible threshold when not using the Linear No Threshold model is not considered as a random variable. Instead results using different models will be presented and this is a good point.

Comments on the executive summary

This part of the report is well done. Several comments were made during the December 2011 phone conference and it is understood that they will be taken into account.

Individual Input from Peer Review Committee Members

Jeff R. Gabor – ERIN Engineering and Research, Inc.

Summary

The State-of-the-Art Reactor Consequence Analysis (SOARCA) project has applied modern analysis tools and advanced methodologies to assess the potential consequences from selected hypothetical severe reactor accidents. The SOARCA project is a significant step forward in severe accident consequence analysis which in the future will provide valuable input to risk assessments. These risk assessments that support the operation of current reactors and the licensing of new reactors must be based on best-estimate evaluations and not unduly biased by conservative assumptions. The SOARCA project objectives are stated as:

- Develop a body of knowledge regarding the realistic outcomes of severe reactor accidents
- Incorporate significant plant improvements and updates not reflected in earlier assessments
- Evaluate benefits of mitigation improvements
- Enable NRC to communicate severe-accident–related aspects of nuclear safety to stakeholders
- Update quantification of offsite consequences found in NUREG/CR-2239

The independent Peer Review Team that was formed includes experts in all phases of severe accident analysis. The majority of my comments on the SOARCA project have been focused on severe accident progression and radionuclide release. My attention has been applied to the use of the MELCOR code in modeling the plant response to severe accident conditions and any modeling assumptions used in the evaluation. From my past experience with a significant number of severe accident analyses, the SOARCA accident progression analysis work represents an advancement of the state-of-the-art in severe accident analysis. The accident progression analysis is thorough and addresses the key severe accident phenomena identified by experts throughout the world. The evaluation makes excellent use of available experimental evidence from a vast array of international programs. Where it is true that the details of any such study are dependent on the specific plant and scenarios being evaluated, the methods and underlining modeling techniques applied in the SOARCA accident progression analysis could apply to any LWR .

Overall, SOARCA successfully addressed the major objectives of the project related to severe accident progression by using state-of-the-art deterministic methods for modeling severe accident plant response. However, 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.

It should be noted that the focus on individual accident progression scenarios in a deterministic framework has limitations. As identified in my specific comments below, the consequences of specific severe accident scenarios can be strongly influenced by the selection of the accident progression paths. While the SOARCA team focused primarily on the important (or more likely) path, the consequences computed are a strong function of the path selected. This is why the presentation of risks must be made in a fully probabilistic framework, rather than a quasi-probabilistic framework like the one adopted by the SOARCA project. 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.

The original consequence analyses portrayed in NUREG/CR-2239 preceded the NRC's adoption of a Severe Accident Policy Statement and PRA Policy Statement, both of which encourage the staff to adopt a risk perspective in considering severe accidents. While SOARCA has advanced the understanding of severe accident progression and provides representative results for selected severe accident scenarios, it is unfortunate that it was beyond the scope of the project to provide a complete set of results in the context of an integrated risk perspective.

The following sections outline more specific observations and comments associated with my individual review. These comments reflect over 2 years of interactions between the Peer Review group and the SOARCA team. During that period there have been numerous presentations made to clarify issues and to reply to individual Peer Review comments.

Peer Review Assessment

The starting point for accident progression analysis is the selection of the representative sequences that could lead to severe accident conditions. The SOARCA development team utilized a screening technique to identify those sequences with the highest likelihood to lead to core damage conditions and to result in a significant release to the environment for the specific plants being studied and for the limited scope of severe accident scenarios considered. My initial comments related to sequence selection were focused around demonstrating completeness in the study. The current executive summary adequately describes the sequence screening criteria and explains how this method is capable of capturing the most significant contributors to offsite consequences. Where more traditional Level 1 PRA techniques can identify a wider range of sequences and provide additional insights, the SOARCA screening methods are judged to adequately capture the major contributors to off-site consequences for the plants analyzed.

The accident progression analysis represents a state-of-the-art deterministic evaluation and makes significant use of available experimental programs. Several of my initial comments on the accident analysis are provided here along with any resolution provided by the SOARCA development team.

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.

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 5.5 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 the analysis of the MSL creep rupture, the larger radionuclide release results from an earlier challenge to containment. 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.

Hydrogen ignition in SBO - comments were provided to identify the source for hydrogen ignition in the station blackout sequences. Section 5.2.3 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 5.2.3 was revised to include this sensitivity. In addition, the SOARCA results demonstrate that operation of the containment sprays provides for significant scrubbing of the airborne fission products, and therefore limits any possible release should the containment fail as a result of deflagration or detonation of hydrogen.

Uncertainty Analysis – as a follow on to the original SOARCA project, a detailed uncertainty analysis (UA) was developed with input from the Peer Review group. The UA addressed parameters relating to the sequence of events, in-vessel accident progression, ex-vessel accident progression, containment behavior, chemical forms of iodine and cesium, aerosol deposition, and numerous items impacting the off-site dose calculations. The mean values, upper and lower bounds, and the overall parameter distributions were presented to the Peer Review group allowing for significant technical discussion. Overall, the UA approach was found to represent the state-of-the-art and would capture the major contributors to high consequences. Selected items were found that were not included and recommended for inclusion in the final UA document. These items included consideration of; 1) delays in key operator actions, 2) transverse in-core probe failures, 3) SRV tail-pipe failures and, 4) additional inputs affecting Main Steam Line creep rupture.

There was a considerable amount of discussion relating to accident progression on several other topics, however, the items mentioned above were judged to potentially have the most significant impact on the consequence analysis and reflect the great care that is needed in characterizing the comprehensiveness and applicability of the SOARCA results.

Conclusion

This review specifically addressed severe accident progression and radionuclide release. I reviewed the SOARCA documentation based on over 25 years experience with similar accident analyses and primarily looked to answer the following 5 questions:

1. Did SOARCA address the important accident progression phenomena?
2. Does the analysis represent a best-estimate approach making use of available experimental data?
3. Does the study adequately address the uncertainty in severe accident phenomena?
4. Does the SOARCA modeling represent an integrated approach by accounting for the interactions between the primary system, containment, secondary buildings, mitigation systems, and related phenomenology?
5. Does the documentation accurately reflect the analysis performed?

As a result of my review of the documentation and through extensive interactions with the SOARCA development team, I would judge each of these questions to be adequately addressed in the analysis, with the exception of item 3 which is being addressed as part of a separate program. Specific to each of the questions above, my review concluded the following:

1. Table 4.5.9-3 of the ASME Standard for Probabilistic Risk Assessment (ASME RA-Sb-2005) provides a detailed list of Large Early Release Frequency (LERF) contributors to be considered in the containment performance evaluation of a PRA. This represents one of the most concise lists of Level 2 PRA phenomena that can impact the timing and release of radionuclides in the event of a severe accident. With the exception of items that were screened out due to low frequency (e.g. containment isolation failure, ATWS-induced failure), the other phenomena have been addressed in the SOARCA evaluation. In addition, the IAEA Draft Safety Guide, DS393, on Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Plants includes a similar list in Table 5 identifying key severe accident phenomena. Again, except in cases where the low frequency threshold was exceeded, the key phenomena have been addressed in the SOARCA evaluation. Based on these references and the screening out of lower likelihood contributors, the SOARCA analysis addresses the important accident progression phenomena.
2. The SOARCA evaluation does represent a best-estimate analysis of the limited set of selected severe accident scenarios with focus on the current mitigation capabilities at the plants. In addition, relevant experimental results relating to severe accident progression appear to have been reviewed and applied to the overall modeling of the plant.
3. Given the substantial uncertainties in severe accident progression analysis, it is not sufficient to characterize the potential consequences of a severe accident scenario using a single accident progression analysis, even if it is felt to be the best estimate case. As

demonstrated by the sensitivity studies requested by the peer review team, accident progression can be strongly influenced by assumptions regarding potentially beneficial failures (e.g., SRV sticking open). A one-at-a-time sensitivity analysis can demonstrate the robustness of the analysis and also identify critical modeling assumptions and inputs. As part of the SOARCA project and as a result of comments provided by this Peer Review group, several sensitivity analyses were performed and provide a better understanding of the controlling phenomena and identified areas for potential future investigations. These sensitivities were performed in a one-at-a-time manner, which is helpful, but they fall short of addressing all potential outcomes. A full appreciation of the results and uncertainties can only be accommodated in a fully probabilistic assessment addressing the applicable aleatory and epistemic uncertainties. Where the Peer Review group had considerable input into the uncertainty analysis approach, final results were not available to the group prior to finalization of this report.

4. Dating back to the original Individual Plant Examinations (IPE), the industry and the NRC have observed the importance of performing a fully integrated analysis. For example, the interaction between fission product transport and the thermal-hydraulic conditions can be shown to provide a dominant feedback when calculating the source term release to the environment. The use of MELCOR to model all important phenomena and system interactions applicable to the selected severe accident progression scenarios evaluated has provided a more realistic analysis.
5. The SOARCA documentation provides a clear picture of the major assumptions and methodology used to perform the analysis. The executive summary adequately provides the overall conclusions of the analysis with the appropriate details contained in separate appendices.

SOARCA represents a major advancement in our understanding of severe accident progression and radionuclide release. Through the adoption of a risk-informed regulatory environment, severe accident response has become a significant consideration for operating reactors. It will be important that this technology be applied beyond just the confines of the research departments and can be used to provide needed input to risk-informed regulatory decision-making. To this end, it is important that the largely deterministic analytical techniques employed in the SOARCA project be extended into true risk frameworks (i.e., a Level 2/3 PRA) in order to more completely characterize the results and communicate risks.

Robert E. Henry Final Review Comments on the SOARCA Report

Since my background is severe accident phenomenology, my contribution to the SOARCA Review Committee, has been to focus on the MELCOR calculations that were performed for the Peach Bottom and Surry reference plants. This involves the timing of events related to core damage, RCS challenges and challenges to the containment integrity, as well as fission release, transport and deposition within the RCS and containment, including the possible releases to the environment.

In my original review of the SOARCA process and the preliminary results, I made the following two comments.

1. The SOARCA Program is a major step forward in developing a credible, integral, technical basis for evaluating the consequences of possible radiological releases, that carries forward all of the lessons that have been learned from industrial experience, as well as large scale international experiments and analyses.

2. 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.

As the SOARCA activity approaches its conclusion, I continue to believe that this activity is a major step forward in developing credible, integral analyses for severe accident sequences to be used in regulatory decision-making. Furthermore, I also think that a review of the MELCOR "best practices" document is an important component of the documentation that should be reviewed along with each MELCOR accident analysis that is used in specific regulatory decisions. It is overly-ambitious to expect that integral accident analysis computer codes correctly represent every nuance of an accident sequence; nor is this always necessary. There are numerous times were the shortcomings of an analytical model result in differences between calculations and reality that do not result in any substantive difference in the decisions that need to be made. Conversely, there are other times when the differences could potentially make a difference, if not now, perhaps in future evaluations/decisions. Therefore, it is worth noting where these subtle differences may exist and what physical phenomena may be contributing to a somewhat different response. These are indicated below.

<u>Accident Sequence Selection</u>

The focus on the sequences of Station Blackouts and containment bypass is appropriate since these are the major contributors to risk. How such sequences could be initiated, and the

frequency of the initiation, is site specific. Nonetheless, focusing on these sequences also challenges all involved to conceptualize one, or more, strategies that utilized the full capabilities of the reactor site to keep the reactor core covered with water. As has been fully documented, the experience during the TMI-2 accident was that a small amount of radioactive material was passed outside of the containment due to the use of the “letdown” system. This also generated some confusion at the time, but the SAMGs that were put in place after that accident identified the various ways in which containment isolation could be challenged and guidelines were developed that would make the control room operators cognizant of how to address such conditions. The same insights should be sought-out as more information becomes available for the Fukushima core damage events.

With respect to what is known to date for the Fukushima accidents, the selection of the SBO sequences for the SOARCA is certainly validated. Equally important is the role of RCIC turbine driven injection system, that is conservatively modeled in the SOARCA accident response evaluations, which was found to be extremely effective in the units where it was part of the design (Units 2 and 3). On both units it was able to run long after the batteries were anticipated to be exhausted. These insights will need to be continually reviewed as more information is released by TEPCO to insure that the conditions which existed during the accidents are clearly understood and interpreted.

MELCOR Modeling of the Severe Accident Sequences

The MELCOR computer code provides an integral representation of core damage events for both BWR and PWR sequences and has been applied to the Peach Bottom and Surry reference plants. Those features that are of particular importance to the evaluations for the accident sequences and the references are:

1. The timing of when the top of the reactor core is uncovered,
2. The extent of hydrogen generation as the core is overheated in steam,
3. Release of fission products from the fuel pins into the Reactor Coolant System (RCS),
4. The downward relocation of molten core as the constituent materials melt or are liquefied,
5. The possible challenges to the integrity of the RCS pressure boundary,
6. Relocation of molten core materials into the Reactor Pressure Vessel (RPV) lower plenum,
7. The potential for rapid steam generation in the RCS,
8. Failure of the RPV that would be sufficient to discharge molten fuel to the containment,
9. Release of fission products from the RCS into the containment,
10. The potential for High Pressure Melt Ejection (HPME),
11. The potential for rapid steam generation in the containment,
12. The potential for Molten-Core-Concrete-Integrations (MCCI),
13. The potential for hydrogen burns,
14. Challenges to the containment integrity and

15. The potential for containment by-pass.

Of these, I find the MELCOR results to be reasonable and benchmarked with the appropriate phenomenological experiments, except for some aspects of items #5, #8 and #9. The possible role of each and how they may influence future evaluations are discussed below.

Item #5

Item #5 relates to the possible challenges to the RCS pressure boundary integrity and this includes the in-core instruments that are part of the reference plant designs. The instrument thimbles in both of the designs are fabricated from stainless steel and Inconel tubing and have a central channel that is open to the containment atmosphere. As a result, these materials have the lowest melting temperatures of the materials that form the pressure boundaries in the reactor core. Consequently, these instrument thimbles will heat-up along with the other core materials and as they approach their melting temperatures, they can be anticipated to fail and release fission products and hydrogen to the containment atmosphere. This is consistent with multiple observations from the TMI-2 event (Henry, 2011). These flow paths were not modeled in the MELCOR analyses of the reference plants. Also, complementary scoping analyses were performed by the SOARCA team, assuming no changes in the flow path geometry (no ablation), to quantify the extent of this release for the Peach Bottom accident conditions concluded that these flow paths would result in only a small release to the containment (drywell). Within the context of what was assumed, I agree with the evaluations performed by the SOARCA team. However, the results from the TMI-2 VIP (Neimark and Hins, 1991 and Wolf et al, 1993), showed that the flow paths were dramatically altered. Therefore, the scoping calculations presented to the Review Committee are useful, but do not necessarily provide the total story.

There are two facets of these release paths that should be followed in future integral evaluations. The first relates to open lattice core PWR assessments and specifically to the evaluations of the natural circulation flows from the core to the steam generators. Failures of the instrument thimbles, like those in the Surry design, would occur when the core-to-upper plenum natural circulation flows are becoming influential. Opening flow paths to the containment at this time would cause an outflow of steam and hydrogen from the core and decrease the flows to the steam generators. This could substantially change the evaluations related to the thermally induced SGTR. It is noted that since the flow paths from the core would be to the containment, the long term release of fission products to the environment would not be substantially affected.

The second influence relates to the BWR (Peach Bottom) calculations. Failure of the in-core instruments would result in fission product releases to the drywell and these could subsequently be discharged to the environment if drywell venting were to be implemented or if the containment would be impaired. Possible surrogates for this discharge of steam, hydrogen and fission products to the drywell could be the creep failure of a main steam line upstream of the inboard MSIV or a stuck open safety valve that discharges to the drywell. Of particular note for this flow path is that the fission product vapors and aerosols may not experience extensive scrubbing in the pressure suppression pool. Since the dominant sequences in the SOARCA

analyses experience significant MCCI and significant releases to the drywell, I do not believe that, in the overall sense, this flow path has a major influence on the SOARCA results. Nevertheless, future analyses for these, and other, severe accident sequences should include these flow paths to remove such questions.

Item #8

Thermal attack of the RPV by relocating molten material involves the competitive processes of (1) heat transfer to the vessel wall, (2) heat transfer to the lower plenum structures including those that penetrate the RPV wall and (3) the structural response of the wall and the penetrations. Hence, assessing the nature of the RPV failure is difficult.

The MELCOR analyses neglects the response of the lower head penetrations and evaluates the RPV failure based on the creep response of the lower head only. Because the failure condition is eventually calculated for the sequences considered, the nature of the RPV failure does not have a significant influence on the ultimate fission product releases to the environment and the off-site consequences. However, future analyses may have a somewhat different focus that is more accident management related and these may have a greater interest in how the failure may occur. In particular, assessment of accident management strategies may be impacted if there would be a number of comparatively small holes (failures) in the RPV wall.

Item #9

This item is influenced by Item #5 and was discussed previously.

MELCOR Best Practices Document

As noted above, the MELCOR “best practices” document is an important aspect of the document for the Peach Bottom and Surry analyses, particularly the benchmarks of the MELCOR models with the key experiments that relate to the thermal hydraulic and fission product behavioral characteristics of the reactor core, the RCS and the containment. The benchmarks were presented during the last committee meeting and, it was agreed that these addressed the experiments of particular importance for the SOARCA studies.

Nevertheless, it is worth noting for future analyses that it would be advantageous for future studies if one facet would be added to the MELCOR “best practices”. Specifically, each benchmark results in certain insights that are generally carried forward to the application of the integral system code to reactor systems. Perhaps the most involved of these are those insights that are tied to the nodalization of the core, the RCS and/or the containment; particularly when these relate to the possible formation of natural circulation flows and/or stratification. Some of the containment analyses presented to the Review Committee demonstrated results where natural circulation flows would have been initiated and the calculated temperature distribution would have been modified substantially. These were discussed extensively and it was noted by the SOARCA team that the necessary countercurrent flow model in the containment was not part of the archived code used for the reference plant analyses. Furthermore, it was agreed that, while the calculated results were not realistic, the influence of the natural circulation flows that would modify the calculated results would not substantially alter the fission product releases

to the environment. However, it is recommended that the MELCOR "best practices" include the insights developed from an individual benchmark such that these would be implemented in integral plant analyses going forward as noted in my comment #2 at the beginning of my comments. Without such a procedural step, it is not clear what is learned and used from the benchmarks.

References

Henry, R. E., 2011, TMI-2: An Event in Accident Management for Light-Water-Moderated Reactors, published by the American Nuclear Society, LaGrange Park, Ill.

Neimark, L. and Hins, A. G., 1991, "Metallographic and SEM Examinations of Nozzle Segments," OECD TMI-VIP Program Review Meeting, Argonne National Laboratory, September 23-24.

Wolf, J. R., et al, 1993, "TMI-2 Vessel Investigation Project Integration Report," NUREG/CR-6197.

Review Comments on the SOARCA Project Documents
Emergency Response Modeling

To: Karen Vierow, Chair
SOARCA Peer Review Committee

From: Roger B. Kowieski, P.E.
NTHMC, Inc.
Member of SOARCA Peer Review Committee

Date: January 19, 2012

Subject: **Review Comments on the SOARCA Project Documents Emergency Response Modeling Sections**

My evaluation of the SOARCA project documents mainly focused on the off-site emergency response sections, where I feel qualified to express an opinion.

In my March 30, 2010 memorandum to you, I provided my evaluation and review comments relating to the Surry and Peach Bottom nuclear power plants. Based on my review of the SOARCA project documents, I found that the parameters used and assumptions made in the emergency response modeling were reasonable and adequate. Furthermore, the emergency response timelines used in the modeling are consistent with the actual response action timelines by the off-site response organizations (ORO's), which were observed and documented by me, in the previous exercises at the Surry and Peach Bottom nuclear power plant sites.

Finally, all of my initial review comments on the draft SOARCA documents were satisfactorily addressed in the revised version. Please note that my previous comments and subsequent resolutions are attached, and are an integral part of this memorandum.

Thank you again for the opportunity to serve on the SOARCA Peer Review Committee.

Enclosures

Natural and Technological Hazards
Management Consulting, Inc.

Memo

To: Karen Vierow, Chair
SOARCA Peer Review Committee

From: Roger B. Kowieski, P.E.
Member, SOARCA Peer Review Committee

Date: March 30, 2010

Subject: Review Comments of the SOARCA NUREG Documents with Respect to Emergency Response Modeling

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OVERVIEW

In my review of the SOARCA documents, I mainly concentrated on the Emergency Response Sections as they related to the Surry and Peach Bottom nuclear power plants. For each site, the modeling was performed for six (6) cohorts, which were established for each population subgroup, representing a meaningful number of individuals. The population data were obtained from the U.S. Census Bureau from the 2000 census data. The population was projected to 2005 using a multiplier of 1.053, also obtained from the Census Bureau.

The WinMACCS network evaluation application was used in the modeling, which accounts for site-specific travel direction and speed. For both plants, the travel direction and speed parameters were derived from the Evacuation Time Estimates (ETEs) prepared by each utility, as required by 10CFR50.47, Appendix E. The SOARCA project used a normal weather weekday scenario that includes schools in session. The SOARCA documents correctly state that the Off-site Response Organizations (OROs) generally do not develop detailed protective action plans for areas beyond the 10-mile Emergency Planning Zone (EPZ). For the 50-mile Ingestion Exposure Pathway, the states with support from the Federal Government are responsible for taking protective actions in the event that an incident causes the contamination of human food or animals' feed. The Protective Action Guides (PAGs) are published in the EPA Manual of Protective Actions for Nuclear Incidents, EPA 400-R-92-001, dated October 1991.

The emergency response timelines presented for both plants identified the following:

- Notification of emergency classification levels to the ORO
- Actions taken by the state and local organization such as the siren sounding and broadcast of an Emergency Alert System message
- Evacuation times for six (6) cohorts of population

Based on my participation and evaluation of several exercises at the Surry and Peach Bottom sites, I concur with the response timelines used in the SOARCA emergency response

modeling. The emergency response timelines used in modeling are consistent with the actual response action times observed and documented in the previous exercises.

In my initial review of the draft SOARCA documents, I have made several comments which were satisfactorily addressed in the revised SOARCA documents, Rev. 1-2/15/2010. Details of my comments and subsequent resolutions are provided in the attached two (2) tables.

I appreciate the opportunity to serve on the SOARCA Peer Review Committee.

Comments on Emergency Response Sections by Roger B. Kowieski
And Subsequent Resolution of Those Comments

Peach Bottom SOARCA Document

No.	Peer Reviewer Comments	Response/Resolution	Peer Reviewer Evaluation of Response Resolution
1.	Why is siren used at particular points? It gives the impression that people move at this time. Suggest changing to "siren + ES message."	The figures and associated text describing evacuation timing have been updated to clarify population motion.	The revised figures and text now correctly reflect the Alert and Notification sequence.
2.	Reconsider the 1 hour allowed to evacuate after second siren. (SOARCA team requested feedback from the committee on this 1-hour time.) Peach Bottom long term station blackout.	The data available to the SOARCA analysis team are consistent with the time lines provided in the documentation to within 15 minutes. 1 hour is also standard in evacuation time estimates. Sensitivity study #3 was performed, which includes a delay of an additional 30 minutes in the response of the public. This delay did not result in any changes in the off-site consequences relative to the baseline case.	Sensitivity study (analyses) satisfied the reviewer's comment.
3.	The evacuation time of the Special Facilities is late and will not go over well with the public.	The relevant text has been updated to clarify that these groups shelter earlier in the event and then evacuate at the time specified.	The revised text clearly states that the sheltering is valuable protective action for the Special Facilities in the early stages of the nuclear power plant incident, prior to an evacuation.
4.	It appears that the existing documents do not address the notification of public in case of a siren failure.	Data have been added to section 6.2.5 justifying the assumption that sirens operate correctly.	The sirens operability records show that that the Peach Bottom sirens are 99.8% reliable.
5.	The seismic analysis time line suggests that after declaration of GE by the plant, sirens and EAS message could be activated within 45 minutes. Based on the actual field experience, it takes approximately 15 minutes for the nuclear power plant to notify the state authorities and may take an additional 38-40 minutes before the sirens' activation and EAS message are completed. Therefore, total time required to complete the A/N sequence may vary between 53-55 minutes.	The timelines used in the analyses are very near the times experienced in exercises. To address any difference in timing, Sensitivity #3 was performed, increasing the initial delay in the notification of the public by 30 minutes.	The sensitivity analysis properly incorporated the timelines experienced during the actual exercise events. The results of the sensitivity analysis are reasonable and acceptable.

Comments on Emergency Response Sections by Roger B. Kowieski
And Subsequent Resolution of Those Comments

Surry SOARCA Document

No.	Peer Reviewer Comments	Response/Resolution	Peer Reviewer Evaluation of Response Resolution
1.	One of the accident progression time lines suggests that after declaration of GE by the plant, sirens and EAS message could be activated within 45 minutes. Based on the actual field experience, it could take up to 60 minutes to complete the A/N sequence (Sirens/EAS message).	The timelines used in the analyses are very nearly the times experienced in exercises. To address any difference in timing, Sensitivity #3 was performed, increasing the initial delay in the notification of the public by 30 minutes.	The sensitivity analysis properly incorporated the timelines experienced during the actual exercise events. The results of the sensitivity analysis are reasonable and acceptable.
2.	It appears that the existing documents do not address the notification of public in case of siren(s) failure. Should a siren fail, it may take additional 45 minutes to notify the affected public by Route Alerting procedures.	The siren operating rates were reviewed under the reactor operations program (ROP) and found to be 99.9% at Surry, which would correspond to the loss of about 1 siren. Route alerting for this one area would not affect the total evacuation time of the public. Text has been added to section 6.2.5 to reflect the performance of the sirens.	The sirens operability records show that the Peach Bottom sirens are 99.9% reliable.
3.	There is a strong precedent for presenting only out to 50 miles of data. Consider not showing the 100-mile data. (Bixler 1st pres. Slide 18)	Results in older studies went out to much longer distances: 500 mi in the citing study and 1000 mi in NUREG-1150. SOARCA takes a dramatic departure from these earlier works by limiting consequence analysis results to much shorter distances. The final determination by the NRC staff is to limit the consequence predictions to a 50 mile radius, which is reflected in revision 1 and subsequent revisions of the documentation.	The final determination by the NRC staff to limit the consequence prediction to a 50-mile radius is reasonable and considered to be adequate. The current planning for the ingestion exposure EPZ is limited to about 50 miles from the power plant, because the contamination will not exceed the Protective Action Guides (PAGs) published by EPA and FDA. It is estimated that much of the particulate material in the radioactive plume would have been deposited on the ground within about 50 miles from the nuclear power plant.
4.	The evacuation time of the Special Facilities is late and will not go over well with the public. (Bixler 1st pres. Slide 20)	The relevant text has been updated to clarify that these groups shelter earlier in the event and then evacuate at the time specified.	The revised text clearly states that the sheltering is valuable protective action for the Special Facilities in the early stages of the nuclear power plant incident, prior to an evacuation.
5.	Too much time is spent on the non-evacuating public.	Consequence results for the nonevacuating cohort will continue to be included in the overall consequence calculations, but a short paragraph has been inserted to describe the fraction of the emergency phase risk within 10 miles of the plant that is attributed to the nonevacuating cohort. In some of the slowly developing sequences, 100% of the emergency phase risk is from nonevacuees.	If the nonevacuating public is properly informed and elects not to follow the recommendations to evacuate by public officials, they should be solely responsible for any negative consequences.

Individual Input on SOARCA Report,, Revision 1

David Leaver

Janaury 19, 2012

This note is to record my overall impressions of the SOARCA project and associated documentation. As a peer reviewer, I have had the benefit of reviewing drafts of the four volume report (a July, 2009 draft, a revised draft issued in February, 2010, and a further revision issued near the end of calendar 2011). There were also a number of meetings, all of which I attended, where SOARCA team members (NRC staff and Sandia contractors) presented information developed in the SOARCA project. As part of the peer review process, I and other peer reviewers prepared a number of written comments on the draft documents which are provided, along with the NRC resolution, in the appendices to this peer review report.

There is also to be an uncertainty analysis report in the suite of SOARCA documents. The methodology used in the uncertainty analysis was discussed in a peer review meeting and comments on this methodology were generated by the peer review team. Peer reviewers also had the benefit of reviewing a draft report on uncertainty results.

The SOARCA project and peer review were near completion at the time of the Fukushima accident (March 11, 2011). As a result of the accident and the need for the nuclear community to focus on understanding the accident and developing insights, SOARCA efforts were put on hold for a period of time. This resulted in some delays and additional work to prepare an appendix which compared and contrasted the Fukushima accident with the SOARCA study.

In preparing this note on my overall impressions of SOARCA, I have not repeated my written comments which were submitted as described above. Rather, this note provides my general assessment of the quality and completeness of the SOARCA effort, and presents some broad observations on reactor safety and public health risks associated with operation of U.S. commercial reactors in light of what has been learned from SOARCA.

My overall impression of the SOARCA project and associated documentation is that it is a substantive, high quality effort which makes a significant contribution to the understanding of U.S. commercial reactor risk. In particular:

1. The technical quality of the SOARCA work is high and in my view it provides a major advancement in the state-of-the-art of characterization of integrated severe accident risk in Level 2 and Level 3. In addition to the fact that NRC had access to the resources necessary for such a multi-year, substantive effort (funding, skilled and experienced personnel, peer review resources), the high quality is the result of a number of things that were done leading up to and during the SOARCA project, including:

a. Improved computational analysis tools (an updated version of MELCOR including, for example, validation against recent experimental data on fission product release; a new, Windows-based version of MACCS2, WinMACCS); methodical consideration of choices among alternative modeling options for addressing important, but uncertain aspects of severe accident behavior per the SOARCA volume entitled, "MELCOR Best Modeling Practices")
b. Assessing the impact of severe accident mitigative features and operator actions to mitigate accidents (explicit consideration of such features and operator mitigation actions, developed over the last few years, had not been done in risk assessments prior to SOARCA)
c. Modeling emergency response in a realistic and practical manner using site-specific information and taking advantage of advancements in the consequence model (WinMACCS) which allowed detailed integration of protective actions into consequence analysis, providing significant advancement over previous studies

An additional, non-technical point indirectly supporting the quality of the SOARCA project is the transparency which has been and continues to be a key objective. This is evident from information presented by NRC at Regulatory Information Conferences in the last several years, previous NRC meetings with the ACRS as well as upcoming meetings where the SOARCA documentation will have been made available to ACRS members, an extensive outside peer review (resulting in this peer review report), upcoming public meetings which are being scheduled, and a very complete set of reports to be issued once Commission approval is obtained. It is apparent that full, open communication on SOARCA is an extremely high priority to NRC, to the benefit of all stakeholders.

The internal event Level 1 work, while not advancing the state-of-the-art, utilized the latest Level 1 information available (NRC's plant-specific SPAR models and Surry and Peach Bottom licensee PRAs). In addition, the NRC interfaced closely with the Surry and Peach Bottom plant staffs during development of the Level 1 information, and the plant staffs were asked to review the documents for fact checking.

Regarding external event Level 1 work, while utilizing the best available external events information, the selection process in SOARCA for external event sequences was less clear. SOARCA does acknowledge that detailed sequence characteristics are more difficult to specify for external event scenarios. Because of their potential for risk (as borne out by the Fukushima accident), large seismic events should be assessed as part of a separate, future study integrated into the NRC seismic research program.

2. On the matter of completeness of scope, the SOARCA project has taken an approach that in my judgment is technically sound. In risk assessments completeness is never perfect, and SOARCA does not address every aspect of reactor risk, nor does it purport to. It has, however, identified those classes of accident events which were not considered as part of SOARCA. My judgment is that none of these classes of accident events is likely to substantially alter the SOARCA findings

on reactor risk. However, there would be benefits to applying more detailed best estimate, SOARCA-like methods to at least some of these classes of accident events (multi-unit risk as an example, again as borne out by the Fukushima accident). In addition, it would be beneficial if SOARCA were to be extended to other LWR plant types (e.g., BWR Mark II and PWR ice condenser containments) which would further strengthen the completeness of the effort.

3. On the matter of completeness of sequence selection, the Level 1 (cdf) screening process used in the SOARCA project as part of sequence selection is reasonable from a technical standpoint. Again, while not perfect, in my mind there are several points supporting the SOARCA process and the fact that risk-significant scenarios were not overlooked:
    a. The process was not so much a black and white, above the line-below the line process as it was use of the cdf frequency screens as guidance with intelligence applied in looking below the frequency screens for higher consequence events that could impact risk (in fact, examples were cited where scenarios below the screen would not have consequences high enough to offset the lower frequency).
    b. High consequences in previous risk assessments, such as WASH-1400 and NUREG-1150, were the result of bypass sequences and severe accident phenomena (e.g., steam explosion, direct containment heating, hydrogen detonation) assumed to cause early containment failure. Bypass sequences are explicitly addressed in SOARCA. With respect to severe accident phenomena leading to containment failure, as a result of the investment of significant time and resources in a number of experimental and analytical studies over the last several decades, these phenomena have been shown to be essentially impossible in an LWR severe accident environment.
    c. Mitigative actions not previously considered in risk assessments have a significant effect in mitigating consequences and providing confidence in the risk results.

An additional point is that a full-scope Level 3-oriented process to determine those sequences important to risk would have required a substantially greater commitment of resources than what was done for SOARCA. Having said this, the NRC has announced that over the next four years it will be performing a full scope, site-wide Level 3 PRA including an analysis of multi-unit accidents initiated by internal and external causes during any plant operating mode.

A final point is with regard to the Fukushima accident. As pointed out in SOARCA's Fukushima appendix, while there were differences in timing and other details of accident progression between the Fukushima accident and the SOARCA study, with respect to accident types and offsite health consequences there is nothing forthcoming from the Fukushima accident that suggests that the SOARCA selection process overlooked important accident scenarios.

Some broad observations on reactor safety and public health risks associated with operation of U.S. commercial reactors in light of what has been learned from SOARCA are as follows:

- While it has long been recognized, or at least strongly suspected, within the nuclear safety community that the characterization of commercial LWR risk in previous studies was unrealistic

and excessively conservative, the SOARCA project has now provided very strong, convincing evidence of this. More work remains to be done, but in my view there is little doubt that fission product releases are dramatically smaller and delayed (even without the mitigative measures discussed below) and thus that the associated public health risks are greatly reduced, much lower than perceived in many quarters.

- The 50.54 (hh) mitigative measures, indeed the integrated set of operator actions including EOPs and SAMGs, considered in SOARCA are very important not only because of the risk impact but also because of the fact that these measures provide margin for uncertainties in sequence selection and analysis, and make the SOARCA risk predictions even more robust . These measures were put in place relatively recently and had not been considered in previous risk studies. SOARCA has not attempted to quantify the probability of success of these mitigative measures. A human reliability study addressing the full set of integrated mitigative measures (e.g., incorporates the measures into the SPAR models) would be valuable to provide this quantification and to serve as a metric to assess the effectiveness of the effort to improve the reliability of 50.54 (hh) measures.
- Perhaps the most significant finding from the SOARCA project regarding safety of operating LWRs is the importance of the role of emergency response and accident management in reducing risk. The TMI-2 accident, the Fukushima accident, and the events of September 11, 2001 (did not involve reactors, but could have) show the importance of this, and collectively suggest that even with the benefit of decades of operating experience plus the insights from PRA, it is difficult for the nuclear community to anticipate all possible events which could cause a challenge to safety at a nuclear plant or possibly lead to an accident. Because of this, a diverse, flexible emergency response system is needed which focuses on maintaining key safety functions of core cooling, containment integrity, and spent fuel pool cooling, and for which there is high assurance that the system will exist and be functional under unexpected conditions which could cause the accident in the first place. For operating LWRs, a crucial aspect of this emergency response system is accident management including integrated EOPs, SAMGs, 50.54 (hh) measures, and the equipment and management systems to assure reliable implementation. While SOARCA was (properly so) performed with no agenda with regard to application of the results, it is now an appropriate time for the nuclear community begin consideration of how SOARCA methods and results could be used to further improve LWR safety and provide even better optimization of safety resources, with particular focus on a diverse, flexible, reliable emergency response system.

Comments on Late-2011 Issued SOARCA Documents Including Executive Summary and Fukushima Appendix

David Leaver

January 19, 2012

1. SOARCA has added an appendix to the Main Report on the Fukushima Dai-ichi accident. This appendix in my view has the correct objective (compare and contrast the Fukushima accident and the SOARCA accident for a reasonable list of topics) and is presented at an appropriate level of detail. My main comment is that there is currently no mention of Fukushima in the main body of the Main Report or the Table of Contents. It is suggested that SOARCA add a short paragraph on Fukushima to the introductory portion of the Main Report which acknowledges the Fukushima accident by paraphrasing information from the Objective section of Appendix A and refers the reader to the appendix. It is also suggested that NRC indicate that there are a number of other topics associated with the Fukushima accident which while not discussed in Appendix A are being or will be treated as part of other work to provide more detailed understanding of and insights from the accident.

2. The individual PB and Surry volumes make the following referrals to Fukushima:
    a. Volume 1 (PB) – Mentions Fukushima briefly on page 12 while discussing LTSBO mitigation measures
    b. Volume 1 (PB) - Includes several paragraphs of Fukushima discussion in Section 6.6 in the context of supporting ability of plant and offsite to truncate the release at 48 hours for PB
    c. Volume 2 (Surry) – Includes similar paragraphs of Fukushima discussion in Section 6.6 in the context of supporting ability of plant and offsite to truncate the release at 48 hours for Surry

    This is somewhat uneven and tends to provoke questions as to why there is not more discussion of the Fukushima accident in the plant-specific volumes. A better way to do this would be to place these three referrals in the Fukushima main report appendix, and then mention in the main report introduction (see comment above) that due to time limitations no attempt has been made to include Fukushima information in the plant-specific volumes.

3. There is a need for a consistent presentation and qualitative characterization of the likelihood of success of mitigated vs. unmitigated scenarios in SOARCA. At the December 6-8, 2011 peer review meeting, NRC indicated that, in the absence of a quantitative HRA study, it did not want to take a definitive position in the SOARCA documents that mitigation would succeed. This is understandable.

    On the other hand, at the meeting there certainly appeared to be a sense among the SOARCA project team members that, based on the plant walkdowns and other information gathered by SOARCA on operator response to accidents, mitigation success is more likely than not. In

addition, there are numerous statements in the current draft documents that qualitatively support the reasonableness if not the success of mitigation:

a. Main Report, page 30: In reference to the ISLOCA scenario frequency in the licensee PRA, the report states that "This frequency does not include any consideration of averting core damage by refilling or cross-connecting RWSTs. This is a significant conservatism."
b. Main Report, page 78: "…scenarios were analyzed including an assessment of reasonable mitigation measures for which procedures and equipment (and training) exist."
c. Main Report, page 78: "The SOARCA assessment and analyses demonstrate the feasibility and benefits of B.5.b mitigation for the analyzed scenarios."
d. Volume 1, page 19: "The specific actions necessary to accomplish local, manual startup and operation of RCIC are delineated in plant procedures, and the actions are reviewed as part of routine operator training. Therefore, successful RCIC blackstart is assumed to occur in the baseline calculation for the STSBO."
e. Volume 1, page 207: "It is expected that mitigative actions would be attempted and that unmitigated variants are less likely."
f. Volume 1, page 219: "The staff expects that plant actions would be successful in mitigating the important severe accident scenarios considered in this study."
g. Volume 2, page 93: The following statement introduces the Section 5 integrated analyses, and states that the unmitigated scenarios are sensitivity studies with the implication that the unmitigated versions are not the expected outcomes but are done for completeness and to provide a basis for comparison with previous studies.

    "The analysis includes calculations to confirm the table-top exercise results to ensure that the timing and capacity of mitigation measures are sufficient to prevent core damage or delay or reduce fission product releases. This analysis also includes sensitivity calculations without B.5.b mitigation measures."

h. Volume 2, page 232: In the context of the MELCOR analysis of unmitigated ISLOCA, it is stated that, "…assuming failure to refill or cross-connect RWSTs for 13 hours is a significant conservatism."

A suggested way to provide this consistent presentation and qualitative characterization of the likelihood of mitigation success is to characterize all of the unmitigated scenarios as sensitivity studies in Section 5 of the two plant volumes (i.e., implement what is stated in the introduction to Volume 2, Section 5). That is, for PB LTSBO, Surry LTSBO, Surry STSBO, Surry STSBO with thermally-induced tube rupture, and Surry ISLOCA, present the mitigated version first followed by the sensitivity study (unmitigated version) as is done for PB STSBO and Surry Spontaneous SGTR.

4. Sections 6 and 7 of Volumes 1 and 2 should be changed to be consistent with the presentation and qualitative characterization of mitigation success described in comment 3. As written, Sections 6 and 7 launch into emergency response parameter development and associated consequence analysis for almost entirely unmitigated scenarios with little or no explanation. See, for example, Table 14 in Volume 1 and Table 6-1 in Volume 2 which together list 16 scenarios, 14 of which are unmitigated. I understand the need to have a scenario that challenges EP and produces consequences, but Sections 6 and 7 should walk the reader through the selection of scenarios in a way that is consistent with the consistent presentation and qualitative characterization noted above.

5. The Executive Summary also has a similar problem of optics on mitigation success. The top of page 3 of the Executive Summary mentions that accident scenarios are analyzed both mitigated and unmitigated, but then pretty much the entire discussion on pages 3 and the very visible figures on page 4 address unmitigated results without any mention of the very significant results regarding unmitigated scenarios. Discussion of unmitigated scenario results is somewhat obscured in the middle of page 5. A more even-handed discussion would better represent SOARCA results.

6. The first sentence of the second paragraph of page 5 of the Executive Summary should be expanded to state that SOARCA results demonstrate the benefits of employing not only 50.54 (hh) security enhancements, but also EOPs and SAMGs. For example, RCIC blackstart is part of SAMGs.

7. On page 7 of the Executive Summary, it is suggested that the parenthesis statement be changed to say factor of 100 for unmitigated LTSBO and factor of 3 for unmitigated STSBO and ISLOCA. The current terminology of “smaller releases” and “larger releases” is undefined and confusing.

8. SOARCA should consider stating at the end of Main Report Section 1.7 (or perhaps in the SECY that forwards the SOARCA reports to the Commission) that, based upon the importance of the mitigation measures, the HRA study mentioned on page 19 of the Main Report should be performed, for example as part of the NTTF work on reliability of 50.54(hh) measures or as part of the Level 3 PRA project. Such an HRA study could provide a metric to measure the effectiveness of the effort to improve reliability of 50.54(hh) measures.

9. Despite its best attempts, SOARCA will be criticized by some members of the public for not considering worst case accidents. Much has been done in the original work and report updates to defend the approach which is good. One additional thing which would be useful in this regard is to provide a general statement regarding the effect of using higher percentile weather on the overall conclusions. I would be surprised if 90% or 95% meteorology changed the overall conclusions at all, i.e., still essentially no early fatalities, and still small LCF effects in comparison to NRC Safety Goal and most still controllable.

10. The LCF risk consequence bar charts in Section 7 of the two plant volumes plot cumulative LCF risk vs. distance. As was discussed at length in the December 6-8 peer review meeting, in several

of these charts (Surry unmitigated ISLOCA, PB STSBO w RCIC blackstart, PB STSBO w/o RCIC blackstart) the 20 mile risk exceeds, or is nearly as large as, the 10 mile risk. This presentation is problematic due to the optics of the bar charts. Many readers who open Volumes 1 and/or 2 will quickly turn to Section 7 and look at these bar charts, and will pay little if any attention to the extremely low absolute risks associated with the ordinate values nor to the conservatisms or qualifiers embodied in these charts (e.g., the fact that the charts represent unmitigated accidents, the fact that much of the risk is controllable). The data behind the charts should certainly be presented in the tables, but the charts themselves should be revised to provide some perspective on the LCF risks and put them in context. The fact that the 20 mile risk may exceed the 10 mile risk, once evacuation of the EPZ is taken into account, is not what is important. Rather, what is important is that it is predicted that those closest to the site are able to avoid nearly all exposure by evacuating prior to plume arrival, and the fact that the doses and associated LCF risks out to any distance are so small in comparison to any societal measure of cancer risk (NRC Safety Goal and cancer fatality risks from all causes).

11. There should be some mention of how or to what extent SOARCA QA was accomplished. Page 27 notes that the SPAR models have been developed and maintained under a formal QA program. Can the same be said about the SOARCA models and documentation? If not, then a description of how quality was controlled would be good.

Introduction

This document summarizes my independent technical review of the approach and underlying assumptions and results obtained for the Peach Bottom and Surry SOARCA analyses. My review focused on determining if the assumptions and results are defensible and represent the state-of-the-art. As this reviewer's expertise is related to probabilistic risk assessment (PRA) techniques, my review is limited to the selection and characterization of analyzed scenarios or sequences, and the treatment of mitigation measures and operator actions. Review comments were originally based on the SOARCA Project Report, Revision 1, dated February 15, 2010 and updated based on comment responses dated July 2011 and the updated Revision 4 draft report dated December 23, 2011.

SOARCA Objective

In the Revision 1 draft, the Executive Summary stated, "[t]he overall objective is to develop a body of knowledge regarding the realistic outcomes of severe reactor accidents." The stated supporting objectives are as follows:

1. Incorporate 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 as well as plant updates in the form of power uprates and higher core burnup.
2. Incorporate state-of-the art integrated modeling of severe accident behavior.
3. Evaluate potential benefits of recent security-related mitigation improvements.
4. Enable the NRC to communicate severe-accident-related aspects of nuclear safety to stakeholders including federal, state, and local authorities; licensees; and the general public.
5. Update quantification of offsite consequences of NUREG/CR-2239, "Technical guidance for Siting Criteria Development."

Although these objectives are no longer presented in the Executive Summary, they are stated in a slightly revised form in Section 1.2 of the Main Report. It remains the opinion of this reviewer that these objectives were only partially achieved. This is not to say that the integrated approach to the phenomenological modeling of accident progression was not valuable and that the insight that accident progression proceeds much more slowly than earlier treatments is very informative. However, the innovative and state-of-the-art techniques used in the SOARCA analysis are focused on this phenomenological modeling and are not used for the identification of sequences to be modeled or for the application of security-related mitigation improvements. Although significant improvements were made from the earlier revision of the report, there still remain significant limitations which regards to the selection and frequency quantification of the analyzed sequences. These limitations reduce the ability to effectively communicate severe-accident-related aspects of nuclear safety and limit the ability to provide an update of NUREG/CR-2239.

With respect to Supporting Objective 1, this reviewer would have preferred to see this objective re-written to reflect a more balanced perspective. The change could have been simply to use the term "plant design and operational changes" in lieu of "plant improvements" and to have included consideration of "state-of-knowledge" changes that have occurred in the modeling of accident sequences. These changes would have re-enforced the inclusion of both positive and adverse impacts. In the response to my initial comment on this issue, the SOARCA Project stated that the team attempted to accurately reflect plant conditions and use the latest Level 1 information concerning the initiators derived from the plant-specific SPAR models. They also stated that realistic information such as fuel burnups, power uprates and

contemporary higher population densities, all of which have the effect of increasing negative consequences were accounted for. Therefore, although the language is unfortunate, the practice employed during the execution of the program appears reasonable.

Sequence Selection

In Section 1.6 of Draft Revision 4, it is stated that the selection of important sequences would best be achieved through the use of a full-scope Level 3 PRA. The report notes that this was not feasible because no current full-scope Level 3 PRAs were generally available. Therefore, the SOARCA Project Team chose to draw from available information, In order to demonstrate completeness, a key challenge for the SOARCA project was the identification and selection of the accident sequences to be analyzed.

The approach used for SOARCA was to analyze sequences with a core damage frequency (CDF) of greater than 10^{-6} per reactor-year. In addition, sequences were included that have an inherent potential for higher consequences (and risk), with a lower CDF – those with a frequency greater than 10^{-7} per reactor-year. The report further states that "[b]y adoption of these criteria, we are reasonably assured that the more probable and important core melt sequences will be captured." Draft Revision 1 stated that the sequence identification is consistent with the American Society of Mechanical Engineers' (ASME's) "Standard for Probabilistic Risk Assessment for Nuclear Power Plants," ASME RA-Sb-2005, which defines a significant sequence, in part, as one that individually contributes more than 1 percent to the CDF. The Revision 1 report used an assumed CDF of 10^{-4} per reactor-year to conclude that the SOARCA sequence selection criterion is 1 percent of an acceptable CDF goal and the SOARCA sequences are consistent with Regulatory Guide 1.200 and the ASME standard. Although the ASME Standard remains in the reference list, the discussion of the consistency with the Standard's screening criteria has been removed.

In order to meet the communication and siting objectives, the approach for selecting and screening the accident sequences needs to be defensible and transparent. This reviewer found weaknesses in both. As sequence selection was primarily based on the above screening criteria with some qualitative additions, the approach to screening is directly relevant to the degree at which "the likely (i.e., best estimate) outcomes of a severe accident at a nuclear power plant" were captured and included in the analysis.

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.

It is this reviewer's experience that there are several means that could have been used to limit the scope of sequences addressed by this analysis. These include the following:

1. Evaluate all sequences using simplified consequence techniques and then use the SOARCA techniques for those where the identified consequences are significant. In essence, one refines the analysis based on the significance. This approach has the benefit of ensuring that all sequences are addressed and that those that are significant receive the more detailed and integrated analysis.

2. Map all core damage sequences into consequence groups and analyze the bounding sequence within the group. This approach would again assure complete accountability. The challenge is to be able to identify the bounding sequences. This challenge is avoided by the first approach.
3. Evaluate all significant accident sequences consistent with the expectation of the ASME PRA standard such that their summed percentage is 95% and the individual percentage is 1%. If this approach is performed using CDF, then there is a need to ensure that bypass events are addressed similarly to those proposed by the SOARCA Project. This reviewer believes that the targeted sequences identified in the SOARCA report represent significantly less than the 95% ASME PRA criterion.

A review of the Surry SPAR Model (Version EE.3P) and the Peach Bottom SPAR Model (Version EE-L2-3P) by this reviewer finds an internal events CDF of $6x10^{-6}$ and $3x10^{-6}$ per reactor year, respectively. It would not be unusual to double these frequencies to account for external events, yielding $1.2x10^{-5}$ and $6x10^{-6}$, respectively. Therefore, 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. In response to these concerns, the SOARCA Project Team stated that the selected sequences are representative in terms of reflecting the range of potential consequences and were intentionally not organized to be a comprehensive risk study.

Given the limitation of utilizing available Level 1 risk models, the approach appears to be adequate for demonstrating the benefit of realistic consequence analysis. However, a more satisfying approach would have been to apply state-of-the-art techniques to the developing the accident sequences. Such an approach would have significantly lessened the concern as to the completeness and calculated frequencies of the identified sequences.

Treatment of Mitigation Measures and Operator Actions

A stated SOARCA objective is to evaluate the potential benefits of recent security-related mitigation improvements. However, a stated limitation of SOARCA in Section 1.6 is "a comprehensive human reliability assessment has not been performed to quantify the probabilities of plant personnel succeeding in implementing these measures and the likelihood of success or failure is unknown." The lack of a human reliability assessment severely limits the credibility of the determining the benefit of these additional actions. It also results in incomplete frequency information as the frequencies of the sequences with the added actions cannot be determined. It is this reviewer's opinion that the SOARCA Project did not demonstrate through state-of-the-art techniques that the mitigation improvements objective was achieved.

Update Quantification of Offsite Consequences of NUREG/CR-2239

The off-site quantification contained in NUREG/CR-2239 (the siting study) uses a set of five siting source terms to represent five accident groups, each assigned a suggested representative probability, and states that these values can be adjusted to approximately represent any current LWR design. For example, SST1 was assigned the probability of 1×10^{-5}, the highest consequence and lowest probability of the source terms considered. The siting study used Indian Point as a base case and examined the role of population distribution in determining reactor accident consequences. It also included a sensitivity study using the actual population distribution and 1-year average wind rose from each of the 91 U.S. reactor sites having either an operating license or a construction permit (as existed in 1982). Sensitivity analysis was also performed for reactor size. Although it is valuable to compare the results of SOARCA

such a comparison appears to fall short of the scope needed to meet the objective of updating the quantification of the consequence analysis included in the siting study.

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 process for selecting the scenarios and for applying the security-related recovery actions appears to have considerable limitations. The lack of a Level 3, internal and external events PRA for the selected plants resulted in the use of a sequence selection and screening process that does not guarantee completeness. The failure probabilities for the additional security-related mitigation actions were not determined preventing the determination of the mitigated sequence frequencies and the full assessment of their impact.

And finally, the objective of updating the consequence analysis contained in NUREG/CR-2239 does not appear to be fully achieved. The siting study has much broader scope addressing all the sites. To fully achieve this objective additional analysis is required.

The integrated approach to the phenomenological modeling of accident progression provides valuable insights and demonstrates the importance of using the current body of knowledge regarding the realistic outcomes of severe accidents. It also makes clear that applying similar techniques to the sequence identification and quantification process would likely yield valuable insights and significantly improve our understanding of both the accident progression and offsite consequences.

URS

January 26, 2012

TO: **Professor Karen Vierow**
Associate Professor and Graduate Coordinator
Department of Nuclear Engineering
Texas A&M University
337 Zachry Engineering Center, 3133 TAMU
College Station, TX 77843-3133

FROM: **Kevin O'Kula**

SUBJECT: **Individual Assessment on State-of-the-Art Reactor Consequence Analyses (SOARCA) Project**

Introduction and Summary

A final assessment has been conducted of the State of the Art Reactor Consequence Assessment (SOARCA) Project and is part of the Peer Review Team's review of the methods, inputs, analyses, results and findings for Peach Bottom and Surry Nuclear Power Plants for selected severe accident sequences. The review of the SOARCA project by this peer reviewer was based on presentations and handout made at four of the five Peer Review meetings in Rockville and Bethesda, Maryland,[1] multiple NRC staff-SOARCA team-Peer Review telecons, and a full series of supporting SOARCA documentation.

The SOARCA project analysis is reviewed in the course of the next seven sections as follows.

Section 1 - Adequacy of the SOARCA Concept
Section 2 – SOARCA Approach
Section 3 – Reasonableness of the SOARCA Technical Results
Section 4 – Uncertainty Analysis
Section 5 – Attainment of the SOARCA Objectives
Section 6 – Final Items for Consideration, and
Section 7 – Suggestions on Improving Current Reports.

[1] This reviewer was not able to attend the fifth and final meeting in December 2012.

1. Adequacy of the SOARCA Concept

It is the judgment of this reviewer that the SOARCA study more than met its goal of applying state-of-art, and valid approaches for evaluating severe accident phenomena and ensuing subsequent offsite consequences. With respect to offsite consequence analysis, many parts of the analyses were technically "cutting edge". Included were:

- Use of a high resolution, 64-sector, polar coordinate grid in the atmospheric transport and dispersion (ATD) in MACCS2
- Modeling of emergency phase actions, specifically use of the network evacuation model and accounting for EPZ roads and their capacities
- Improved assessment of shielding factors and assignment of population cohorts
- Capability to input current FGR-13 as well as older sets of dose conversion factors
- Improved, updated ICRP-60 dosimetric models and the capability to run a full range of latent health effect models.

However, other aspects of the offsite consequence analyses maintain older models or input data. These are acceptable for achieving the overall goals on the SOARCA project in most cases but merit some thinking in the direction of upgrades or replacement with other options for later work in commercial plant offsite consequence analysis studies. A list of candidates under this improvement area includes:

- Straight-line Gaussian plume segment model
- Economic consequence model with older (e.g. NUREG-1150, Ref. 2) data, and
- Assumptions and input data associated with decontamination and cleanup of economic assets. As with economic consequences, calibrating this important part of the analysis with the approach taken for decontamination in the mid- to late eighties isn't consistent with a state-of-the–art analysis.

The limitations, were outweighed by the strengths of the SOARCA methods and skills of the Project team. The net outcome was a project that met its goals.

2. SOARCA Approach

SOARCA project, practices and methodologies were described for the most part, with appropriate level of detail for the accident scenario selection process. Criteria for event selection and screening are defined sufficiently and appropriate for the intent of the SOARCA analysis.

For the Peach Bottom plant, this process resulted in the following accident scenarios:

- Long-term station blackout (unmitigated and mitigated)
- Short-term station blackout with RCIC blackstart (unmitigated and mitigated), and
- Short-term station blackout without RCIC blackstart (unmitigated and mitigated).

For the Surry plant, the SOARCA process resulted in the following accident scenarios:

- Long-term station blackout (unmitigated and mitigated)
- Short-term station blackout (unmitigated and mitigated)
- Short-term station blackout with thermally induced steam generator tube rupture (unmitigated and mitigated), and
- Interfacing systems loss of coolant accident (unmitigated and mitigated).

I remarked in 2010 that the discussion presented during peer review meetings and in the documentation was sufficient to justify using a set of robust scenarios for each plant. No additional information has been observed that contradicts this conclusion. More complete modeling and the addition of other scenarios will be best saved for a near-term, full-scope Level 3 analysis, should that objective be chosen by the NRC.

The approach taken for the offsite consequence analysis was comprehensive and met expectations for contemporary standards and assumptions. Innovative methods were applied rigorously for, but not limited to

- Evacuation (network) modeling and cohort representation
- Publishing of latent cancer fatality risk results through three different health effect models
- Highly accurate dispersion polar coordinate grid
- Site-specific dose mitigative setpoint modeling and other aspects of emergency planning analysis.

The interface between the MELCOR source term development and input to the offsite MACCS2 consequence analysis is now better documented than earlier. In other words, while satisfactory MELCOR-MACCS2 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. However, a complete discussion should be documented fully in the near future to allow safety professionals and regulators an opportunity to draw their own conclusions.

Nonetheless, the chronological treatment applied in the SOARCA analysis was notably consistent from scenario selection through offsite consequence evaluation for each of the baseline, accident sequences discussed in the NUREG/CR reports. More will be said about the integration in Section 6.

The SOARCA processes were sufficiently best-estimate with respect to the offsite consequence analysis performed. It will be important to finalize the uncertainty analysis to understand parameter impacts and sensitivities, and where the best-estimate values lie relative to other statistical figures-of-merit.

3. Reasonableness of the SOARCA Technical Results

In general, the overall technical results are well substantiated and explained in sufficient detail so as to support key findings and study insights. Good use was 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, and comparison to the latent cancer fatality quantitative health objective (NRC safety goal). An opportunity was missed to connect with one or two metrics reported for Peach Bottom and Surry from NUREG-1150. A SOARCA study examination of an accident sequence from the 1990 NUREG-1150 study and compared with a SOARCA unmitigated sequence for population dose or mean land contamination would have presented useful insights one step more directly associated with the reduced source term, than obtained by reporting LCF risk metrics. In other words, use of the LCF to provide comparison

was done well but needed to add one last layer of calculation through use of one of three health effect models. This would not have been required if analysis were terminated earlier.

4. Uncertainty Analysis

For credibility with the PRA community, the SOARCA project, as a best-estimate, realistic approach to severe accident sequence analysis, requires an understanding of the implied margins in intermediate quantities and in the reported LCF risk quantities. The peer review panel was informed on the status of the uncertainty analysis and responses to earlier memoranda issued in 2010 during the December 2011 meeting presentation by Dr. Ghosh, the draft Uncertainty analysis document provided in late December, and the last telecon conducted in early January 2011.

While the panel had an opportunity to see this work in progress and ask questions, it was informed that no additional opportunities would be available to comment on this important aspect of the SOARCA Project. Having thought about this over the last few weeks and recognizing the quality of the current draft document ("State-of-the-Art Reactor Consequence Analyses (SOARCA) Project: Uncertainty Analysis"), this is a satisfactory if the final loose ends are completed. I'll note one last time that the single parameter that would have provided additional insights in its incorporation in this type of evaluation is the habitability – long-term dose criterion (DSCRLT). Because this aspect of the analysis, i.e., when residents would be able to return to their homes, results in this long-term dose and this eventually becomes the basis for a small but finite LCF risks in some of the health effect models, it would be of interest to quantify at some point (perhaps at minimum in a sensitivity analysis study).[2]

5. Attainment of SOARCA Objectives

The judgment of this reviewer is that the SOARCA project largely met its over-arching objective as stated in the SOARCA Main Report, (page 10 of NUREG-1935), i.e., "to develop a body of knowledge regarding the realistic outcomes of severe reactor accidents," and "to develop best estimates of the offsite radiological health consequences for potential severe reactors accidents for two pilot plants. Included with the primary objective are corresponding and supporting objectives. While many areas are still being addressed and need additional work, these include: (i) incorporating 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 (10 CFR 50.54(hh)) as well as plant updates in the form of power uprates and higher core burnup; (ii) crediting state-of-the-art integrated modeling of severe accident behavior which includes the insights of some 25 years of research into severe accident phenomenology

[2] Also, after the Fukushima event, the need to plan, deploy, mobilize equipment, and coordinate large-scale decontamination operations would seem to require some intermediate time period to be included. Currently, the model in the SOARCA Project jumps immediately to the long-term activity without an in-between period. A rarely used time phase, the intermediate phase, can be modeled in MACCS2 between SOARCA's seven-day emergency phase and the fifty-year long-term phase. A sensitivity analysis could help examine what level of exposure decrement could be achieved with a preliminary waiting period. This may be left to the next full-scope Level 3 PSA.

and radiation health effects; (iii) evaluating the potential benefits of recent security-related mitigation improvements in preventing core damage and reducing an offsite release should one occur; (iv) enabling the NRC to communicate severe-accident-related aspects of nuclear safety to stakeholders; (v) updating quantification of offsite consequences found in earlier NRC-sponsored work such as NUREG/CR-2239 and NUREG-1150.

6. Final Items for Consideration

Past review reports have itemized as many as eight areas as unaddressed or opportunities for improvement. Through the responsiveness of the SOARCA Project Team, the original list has been addressed to disposition: (1) surface roughness length, (2) deposition velocity, (3) non-site specific and site-specific parameters (partially), (4) boundary weather, and (5) polar grid dimensions. The final documents are readable, sufficiently scrutable and there is a much improved appropriateness of presentation. There are still several items that could be considered before the full set of NUREG documents is published.

1. **MELCOR-to-MACCS2 transition:** The documentation in the NUREG-1935 Main Report, and in the plant-specific NUREG/CR-7110 reports, is incomplete with the MELCOR accident scenario source term to MACCS2 input transition. The pedigree of the MELMACCS technical report (Ref. 49 in the executive summary Main Report) remains an internal laboratory report. It is recommended that the report be formally added to the SOARCA document collection .

2. **Listing of the MACCS2 input parameters used**: Although this is not a full discussion, the input parameter values in the Surry and Peach Bottom appendices is adequate at this point, and much appreciated. In time, a discussion and justification would be key information in defending how the model was run.

3. **Centralized and plant-specific discussion of MACCS2 improvements:** Section 5 of the Main Report summarizes improvements in the Offsite Consequence Analyses, and specifically, those features in Version 2.5 of the MACCS2. This section mentions a number of the major improvements to the original MACCS2 code used to quantify SOARCA offsite consequence predictions. To improve the level of detail in the reports regarding how these improvements were utilized, it is suggested that an additional level of detail be provided in the plant-specific reports, in Section 7, immediately after Surry/Peach Bottom Source Terms.

4. **Reporting of additional risk metrics consequence measures** – In addition to the conditional and absolute health effect risks reported in the SOARCA study, the metrics of population dose in the fifty-mile region and land contamination (I, Te, Ba and Cs would be obvious choices based on the plant-specific results reported. The population dose metric would allow comparison to some selected accident scenarios used in NUREG/CR-2239 and in NUREG-1150.

7. Suggestions on Improving Current Reports

The addition of an appendix to the Main Report on SOARCA and the Fukushima Daiichi Accident is strategically important to the acceptability of the project results as a whole. However, a few lines should be mentioned in the Executive Summary about the statement on Fukushima and its relationship to the SOARCA Project, and acknowledged in the Table of Contents.[3]

The order of discussion in the plant-specific reports appears to be somewhat out of sequence with the order followed in the main report. For example, each plant report transitions from integrated thermal hydraulics, accident progression, and radiological release analysis into emergency preparedness, with a subsequent results section. This is a somewhat sharp transition without the expected phenomenological logic that is carried in the main report.

There is mention in the Main Report on "alternative long-range plume spreading model" as part of one of the atmospheric transport and dispersion modeling improvements. However, there appears to be no additional discussion on this feature in the main report or in the plant-specific documents.

Acknowledgement

As a reviewer of the multi-year SOARCA Project, I wish to thank all the SOARCA Project Team for their presentations, technical interchange, responsiveness and willingness to work with the peer reviewers. In particular, Dr. Nate Bixler is appreciated for his time carefully explaining his position and thinking applied in the SOARCA consequence analysis modeling area.

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4. *State-of-the-Art Reactor Consequence Analyses (SOARCA) Project: Uncertainty Analysis*, Draft Report, NUREG/CR-XXXX, Sandia National Laboratories.

[3] This may have been mentioned or agreed to in the December 2011 meetings or during the phone call in January 2011.

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8. D.I. Chanin, M. I. Young, and J. Randall. *Code Manual for MACCS2: Volume 1, User's Guide*; NUREG/CR-6613 (SAND97-0594), Sandia National Laboratories, published by the U.S. Nuclear Regulatory Commission, Washington, DC, 1998.

Peer Review Comment on the SOARCA Project and the Resultant NUREG-1935 Report

John D. Stevenson

January 23, 2012

This note is to record my peer review of the SOARCA project and associated documentation. As a peer reviewer, I have reviewed drafts of three NUREG's as follows:

- NUREG/CR-7110, "State-of-the-Art Reactor Consequences Analysis Project," Vol. 1, Peach Bottom Integrated Analysis.
- NUREG/CR-7110, "State-of-the-Art Reactor Consequences Analysis Project," Vol. 2, Surry Integrated Analysis.
- NUREG-1935, "State-of-the-Art Reactor Consequences Analysis (SOARCA) Project," Main Report.

There were also a number of peer review meetings which I attended, where SOARCA team members (NRC staff and Sandia contractors) presented information developed as part of the SOARCA project. As part of the peer review process, I prepared a number of written comments on the draft documents listed above and presentations that were made at the peer review meetings which are documented in the appendices to this peer review report.

My peer review was limited to my areas of expertise which is associated with the natural hazard external events and the generation of hydrogen in a Loss of Coolant Accident, LOCA with a potential for early release of significant levels of radiation from the containment structures and systems.

I agree from a natural phenomena hazard standpoint, the greatest threat to the loss of electrical power both from sources outside and inside the two NPP's studied resulting in station blackout is a very low probability of occurrence earthquake (i.e. less than a 10^{-6}/yr probability of exceedence earthquake).

The concern raised is that such a level of earthquake for the Surry NPP could lead to the soil liquification or consolidation leading to displacement that might cause containment penetrations to rupture and thereby lead to early containment failure which was identified in the SOARCA study.

A second potential early release from the containment of the results of a core melt accident caused by a potential station black out evaluated by the SOARCA project, was the potential for hydrogen generated by the postulated core melt to be accumulated in a sufficient quantity to cause a detonation within the

containment structure that could result in pressure loads on the containment structure at or near the ultimate capacity of the containment. Deflagration of hydrogen inside the containment has been postulated and considered in the SOARCA evaluation of the potential for early release of core melt radiation from the containment. The potential for the hydrogen concentrations to build-up to a concentration sufficient for a detonation is controlled by the steam in the containment generated by the core melt acting to prevent hydrogen deflagration or detonation. The activating of the containment spray system which is designed to condense the steam in the containment and thereby reduce containment pressure also eliminates the steam as hydrogen deflagration or detonation suppressing agents.

Generally, it is assumed in the design of PWR containments, because of their large volume, that one or more deflagrations of hydrogen generated in the containment due to core melt will prevent the build-up of the hydrogen concentration to detonation levels. In BWR containment dry wells in U.S. NPP's there are installed filtered vents which can be activated to release containment hydrogen to the atmosphere before a detonation level of hydrogen is allowed to occur.

There is a need to assure that the severe accident management team is aware of the potential for hydrogen build-up in containment to detonation concentration levels and for them to develop administrative procedures or install hardware to preclude this detonation level build-up from occurring.

My overall impression of the SOARCA project and associated documentation is that it is a realistic effort to validate the consequences of a complete loss of cooling to a large commercial BWR and PWR power reactor and as such makes a significant contribution to the understanding of U.S. commercial reactor risk.

Review Comments of the SOARCA Project
Karen Vierow
January 20, 2012

In formulating this review, I prepared a list of key questions that should be answered to evaluate the SOARCA project. Topics and aspects of the SOARCA project for which I feel qualified to comment on are evaluated below. Several of the comments are limited to severe accident modeling and have been qualified as such.

1 Adequacy of the SOARCA Concept

1.1 Is SOARCA a valid approach to evaluating severe accident phenomena and the offsite consequences of reactor severe accidents?

The SOARCA approach for modeling severe accident phenomena is a valid approach because it is a comprehensive and integrated analysis approach applied to selected scenarios that could hypothetically lead to severe accident event sequences. Physics-based deterministic methods and probabilistic risk assessments are combined to take advantage of the best of both approaches in the severe accident analyses.

1.2 Is the SOARCA truly "state-of-the-art"?

SOARCA is state-of-the-art for analysis of severe accident sequences in that the latest version of MELCOR severe accident modeling has been adopted.

MELCOR had previously been compared against other leading severe accident codes in the U.S. by this reviewer and other researchers. Multiple journal articles document comparisons against the MAAP code and/or and the SCDAP/RELAP5 code for scenarios similar to those studied by SOARCA. In particular, the high-pressure natural circulation scenario, studied within SOARCA for the Surry PWR reactor, has been extensively studied in these efforts. The thermal-hydraulic phenomena and major in-vessel severe accident phenomena have been demonstrated to be in good agreement for the three codes. The integral effect of diversified core models in terms of total hydrogen production and total core debris mass slumping into reactor vessel lower head were also shown to be consistent for the three codes.

Version 1.8.6 of the MELCOR code has been used in the SOARCA. The changes from MELCOR 1.8.6 to 2.1 are those needed for "modernization" to a newer FORTRAN version. The MELCOR 2.1 code models have been shown to reproduce the results of MELCOR 1.8.6 version out to machine accuracy. Therefore, version 1.8.6 of MELCOR may be considered state-of-the-art for the current purposes.

1.3 Even if SOARCA is state-of-the-art, is the approach adequate to achieve the goals?

As discussed above, the MELCOR code has been shown to be state-of-the-art, with comparable capabilities as other leading U.S. codes for severe accident analysis. Comparison of severe accident code predictions against experimental and plant data is an essential test of code accuracy that provides additional information on the relative merits of the various severe accident models. MELCOR severe accident models have been validated against a number of separate effects tests and the TMI-2 plant data. The presentation by R. Gauntt at the Dec. 8, 2011 meeting revealed that the code validation has been conducted in a systematic manner to test code simulation capabilities for key phenomena and their interactions. Since many of the key models for the SOARCA have been validated, MELCOR may be considered adequate for severe accident calculations in order to achieve SOARCA goals.

A considerable amount of excessive conservatism in past calculations has been removed by incorporating plant improvements and updates into the assessments. The code now produces results that are more realistic than were previous analyses. The severe accident calculations also include modeling improvements and insights that have been achieved since the earlier calculations were performed.

Some analysis aspects remain that require additional sensitivity studies and uncertainty quantification. First, conservative safety factors have been applied in certain areas where uncertainty remains. Second, where scientific knowledge is insufficient to develop mechanistic models, assumption of values for various parameters must be done.

As recommended in an April 9, 2010 memo from members of the peer review committee to the SOARCA team, uncertainty quantification and sensitivity analysis are essential to the credibility of the SOARCA. The Uncertainty Analysis currently being conducted is an important effort to address this recommendation. The peer review committee was presented on Jan. 5, 2012 with the uncertainty parameters and their respective value distributions adopted for the Uncertainty Analysis. The list is subjective, but it appeared to capture the most significant uncertainty parameters.

2 Reasonableness of the SOARCA Technical Results

The severe accident progression results are reasonable, as reported in the SOARCA documentation. The temporal trends and absolute numbers (such as maximum temperature, pressure, etc.) have been explained within the text. Where significant uncertainties exist, these have been investigated in a conservative manner so that results do not include excessive optimism about nuclear plant safety.

3 Attainment of SOARCA Objectives

The SOARCA objectives are, quoting from Section 1.2 in the Main Report:

The overall objective of the 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 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, which 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 or delaying an offsite release should one occur.*
- *Enable the 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"*

The overall objective has been attained for the reactors evaluated by SOARCA, as evidenced by the reduction of conservatism in the evaluations and the use of plant-specific data, procedures, scenarios, and other information. Each scenario has been investigated in careful detail to assure consistent and reasonable evaluations.

The analysis presented here is for two specific plants, a PWR unit at Surry and a BWR unit at Peach Bottom. Many *insights* have been gained; however, care should be taken in extrapolating results to other plants. Since each unit may have unique operating procedures, mitigation equipment, and the like, differences should be identified before applying the results of the current analyses to other plants.

Regarding the first bulleted goal, the attainment of this goal is clearly demonstrated in the SOARCA document as far as plant improvements and updates. The method for inclusion of power uprates and higher core burnup in the MELCOR analysis was clarified in the SOARCA team's July 2011 response to my comment.

Attainment of the second bulleted goal has been achieved for severe accident analysis, as discussed in item 1.

The third bulleted goal has been documented in Volumes 1 and 2, which present the comparisons of mitigated and unmitigated scenarios. Mitigation steps have significant, positive effects on the event progression and consequence reduction.

Addressing the fourth bulleted goal, the documents are thorough and well-prepared. Members of the public who are willing to invest time and have a familiarity with nuclear and related technologies will be able to understand the SOARCA approach and results as presented in the SOARCA documents. For the general public less familiar with the technologies, a document written in layman's terms is needed. Such documents were mentioned at earlier Peer Review Committee meetings, and it is anticipated that they will be produced and disseminated. This last action is essential to attaining the fourth bulleted goal.

I leave evaluation of the fifth goal to others.

4 Presentation of the SOARCA effort as a "best-estimate" study

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 Volume 2 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.

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 readily to mind. First, conservatism is one method for treating uncertainties. Second, 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.

An alternative suggestion is to perform a calculation in which the conservatisms are removed. 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.

The SOARCA team explained in their July 2011 response that they understand this concern and have identified conservatisms throughout the documents. A concise discussion of conservatisms would be convenient, but the lack thereof does not detract from the quality of the SOARCA project.

5 Appropriateness of Presentation in the SOARCA Documents

5.1 Does the SOARCA appear objective and uninfluenced by licensees or other constituents?

The SOARCA project appears to have been conducted independently from licensees and other constituents. While discussions with utility staff were necessary to obtain the required plant descriptions and other information, the evaluations were performed with codes that may or may not be used by plant personnel and without utility involvement.

Representation of industry, consulting, academia and international research institutes on the Peer Review Committee promotes a fair review of the process and makes possible an adequate and impartial evaluation of the SOARCA.

5.2 Will the public interpret the SOARCA as intended?

Those educated in nuclear and related technologies should find the SOARCA document a detailed and well-prepared presentation of the effort. Emphasis on the objectivity and impartial nature of the effort should be emphasized. Stating the NRC's mission to protect the public's health and examples of where the NRC has denied requests for licenses or other permissions may remind the public that the NRC does not gain by painting a bright picture about the safety of nuclear power plants.

As mentioned earlier, a description of the effort in layman's terms is important when communicating with a large percentage of the population. Particular care is needed with respect to presentation of health effects and to assure the general public that all cohorts have been given adequate consideration. The cohort that voluntarily does not follow evacuation guidance must be clearly noted as being voluntary nonevacuees.

The MELCOR Best Modeling Practices volume provided in earlier NUREG drafts is exceptionally helpful in understanding the philosophy and implementation of models for key phenomena. For many of these calculation aspects, code developers and users may arrive at different approaches. Several important aspects of the severe accident evaluations that would not have been apparent otherwise are explained and therefore could be reviewed for acceptability.

Summary Statement

This review has been performed primarily with respect to the severe accident modeling techniques and results.

The severe accident modeling of SOARCA has been performed with a state-of-the-art code version, MELCOR 1.8.6. The code has been demonstrated to have capabilities with at least the same level of fidelity as other leading severe accident codes in the U.S. Most of the models used in SOARCA have been validated against plant data and separate effects test data.

Some analysis aspects remain that require additional sensitivity studies and uncertainty quantification. This reviewer believes that "closing the loop" on remaining issues via uncertainty quantification and sensitivity analysis will enable achievement of the SOARCA goals for severe accident analysis of the Surry and Peach Bottom plants.

The SOARCA objectives, as stated in the Executive Summary, have been achieved in large part. In particular, a large amount of information regarding severe accident analysis has been developed. The plant-specific analyses of a unit at Surry and at Peach Bottom have provided insights into the behavior of other reactors. Care should be taken in extrapolating the results to other plants. Documentation has been well prepared, although a SOARCA document in layman's terms would also find good use.

A considerable reduction of conservatism has been achieved in the SOARCA analyses. 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. The conservatisms remaining in the calculations should be compiled in a single section in the SOARCA document.

Finally, the SOARCA appears to be objective and uninfluenced by interested parties. The presentation seems appropriate. Particular care should be given to presentation of health effects so that the general public understands that all cohorts have been given adequate consideration.

Examination of Off-Site Consequences of a Severe Reactor Accident

Reviewer: Jacquelyn C. Yanch, PhD

Full Report: December 2010

Preface: January 2012

Preface:

The severe reactor accident at Fukushima in March 2011 allowed a number of SOARCA assumptions and findings to be realistically evaluated. Importantly, we saw that there was sufficient time following the start of the reactor accident for the public to be alerted and for evacuation to proceed. Little to no radiation dose (beyond natural background) was received at the time of the accident. The number of acute health effects was zero and most members of the public will receive essentially no additional dose from the accident until (and unless) they are permitted to return to their homes. This confirms the conclusions of the SOARCA study which found, for all scenarios examined, that substantial radiation dose can be effectively avoided through evasive action on the part of each member of the public.

Evacuation is necessary to avoid the potential health effects of living in an environment contaminated with fission products. However, the accident at Fukushima Daiichi has also illustrated the costs, the trauma, and the disruption associated with dose avoidance through evacuation.

The Fukushima accident contaminated over 1000 km^2 of land. Approximately 100,000 people lost their homes and have been forced to live for many months, and perhaps for many years, in crowded shelters and temporary housing. For some, the loss of home and community will be permanent. Evacuees have lost jobs, families have been separated, farmland has been rendered unusable, shops and factories have had to be abandoned, and societal infrastructure in many villages, towns and communities has been destroyed.

Thus the main consequence of a severe reactor accident is likely not radiation-induced health effects in the population but instead, massive disruption of the lives of tens of thousands of people as they struggle to deal with an environment contaminated with fission products, and an economic burden that extends far beyond the evacuees to the nation's taxpayers at large.

No effects on the population other than radiation-induced health impact are evaluated in the SOARCA study and no attempt has been made to evaluate the consequences of environmental contamination and its impact on human suffering, human health, or the local and national economic consequences of a severe reactor accident. This is a significant and important omission. The study results contain the good news that radiation-induced health effects from a severe reactor accident can be avoided and the accident at Fukushima has shown us that this is indeed the case. However the bad news is that undertaking extensive dose avoidance through long-term evacuation and relocation is associated with enormous drawbacks and these drawbacks are very likely more harmful than the radiation dose being avoided. [This is discussed in detail in my review.] Existing data simply do not support extensive societal disruption to avoid dose-rates as low as 20 mSv/yr. No data demonstrate harm at dose-rates this low (~8 times average natural background levels). On the other hand, the drawbacks of long-term loss of home, community, farms, and industry are all too clear. This trade-off needs to be critically examined.

Examination of Off-Site Consequences of a Severe Reactor Accident

Reviewer: Jacquelyn C. Yanch, PhD

December 2010

Abstract:
The SOARCA study's evaluation of the rate of progression of different accident scenarios, coupled with the anticipated rate of evacuation of the public, reduces, to very low levels, the estimated likelihood of any acute effects of radiation. The health-related impact of an accident then results, almost exclusively, from long-term, low dose-rate irradiation. How much radiation exposure the public receives depends on what dose-rates 'trigger' their relocation and their return home. While these trigger levels are set by individual states, not by the NRC, the SOARCA study brings to light significant problems associated with where these levels are set and the impact they will have on the public as they try to meet these levels. For instance, relocation and return home levels are set below the doses received as part of natural background in several parts of the world, and are also lower than the doses received by many people from diagnostic medical examinations. The strategies in place to avoid these radiation doses following an accident place a considerable burden on members of the public and it is not clear that these efforts are justified in terms of better long-term health. 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.

Examination of Off-Site Consequences of a Severe Reactor Accident

Reviewer: Jacquelyn C. Yanch, PhD

Review Summary:

1. Dose to the public is avoided during an accident but is received upon returning home.
For most of the scenarios addressed in the SOARCA study, the accident proceeds slowly enough that, should it be necessary to give the evacuation order, the public can leave in a timely way so that little to no radiation dose is incurred until the public is permitted to return home. When to return home is determined by return-home dose-limits set by individual states.

2. 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.

3. SOARCA approach to estimating health impact reflects the state-of-the-art.
The strategy for determining the impact of exposure to anthropogenic radiation (assuming a threshold for acute effects, integrating the dose over a 50 year period, assuming cancer is the only impact on long-term health, the use of a DDREF of 2.0, and the application of a common risk factor throughout the entire dose range) is broadly consistent with the approach taken by the scientific field in general and by several national and international agencies and committees.

4. Extensive new data concerning reactors are incorporated in the SOARCA documentation but little new knowledge is available concerning the health impact.
While our ability to quantitatively address the likelihood of a severe reactor accident has improved dramatically over the last few decades, there has been little change in the depth of understanding of the consequences of radiation exposure to people, and we know little more today, about the consequences of living with an elevated dose-rate, than we did 30 years ago.

5. Who bears the burden of responding to the accident?
The burden of minimizing radiation dose is normally borne by the nuclear utility, but once radionuclides are dispersed in the environment this burden shifts to members of the public. The public undertakes the significant upheaval, effort, and financial cost devoted to minimizing their radiation dose. At the present time, however, we do not know what dose-rates we need to avoid and therefore we do not know what dose-avoidance efforts are really justified in terms of <u>actual</u> hazards to our health.

6. The return-home dose limits (set by individual states) are set very low, exacerbating the burden on the public. Even the least conservative return home dose limit is *lower* than the natural background doses in many areas of the world. The criterion used in PA is less than a factor of 2 higher than the average background in the Unites States and is significantly less than the dose received from a single CT exam of the abdomen. In this context, major dose-avoidance strategies such as long-term residential relocation until the return-home dose limit can be met, are unlikely to be in the best interests of the public.

Part A: Review Comments

A.1 Dose to the public results from returning home.

The SOARCA study results predict that dose to the public, for nearly all scenarios considered, will be very low. Evacuation training, experimental testing of evacuation, and experience with natural disasters, coupled with improved understanding of accident progression and knowledge of when, after initiation of the accident, release of radionuclides can be expected, has provided significant assurance that radiation exposure to the public in the direct aftermath of all accidents considered can be kept very low. That is, for most of the scenarios addressed, the accident proceeds slowly enough that, should it be necessary to give the evacuation order, the public can leave in a timely way so that little to no radiation dose is incurred until the public is permitted to return home.

When to return home is a decision made by individual states (not by plant management and not by the NRC). Pennsylvania sets the dose-rate limit at which residents can return home at 5 mSv (500 mrem) per year; Virginia follows the EPA recommendation of 20 mSv (2 rem) in year one and 5 mSv (500 mrem) per year thereafter.

Getting to the low dose-rate stipulated by the return-home dose-limits (RHDLs) requires that the public undergo significant upheaval or undertake significant cost and effort. Time will allow for physical decay of the radionuclides and for the effects of weathering [1] however during this time residents must live away from their homes. Alternatively, decontamination procedures such as scrubbing and/or flushing surfaces; soaking, plowing or removing soil; and removal and replacement of surfaces, etc., [1] can reduce dose-rates, but the cost to decontaminate can be considerable. If decontamination costs are greater than the cost of the land or dwelling then the land is considered condemned. [If land is condemned, no dose is accrued by the residents because they never return.]

Given the ready availability of foodstuffs from outside the area affected by the reactor accident, radiation dose from contaminated food and water can be avoided by prohibiting consumption of local produce, livestock, and water. Therefore, radiation dose from ingested radionuclides is not considered in the SOARCA study. Upon return home, then, the dose is assumed to come primarily through external radiation by gamma-emitters deposited on the ground, specifically the long-lived gamma emitters: ^{134}Cs and ^{137}Cs (2.1 yr and 30 yr half-lives, respectively).

A.2 What is the impact of the return-home dose-rates on human health? We don't really know.

Our understanding of what impact the return home dose-rates will have on people is very primitive. In fact, we have essentially *no* understanding of the potential health consequences of the dose-rates encountered upon returning home. The limited data we do have regarding radiation-induced health effects are highly uncertain and, in addition, are relevant to situations that bear very little resemblance to the conditions reflected by the return-home dose limits (RHDLs). [This is discussed in detail in Part B.]

More important perhaps is the fact that we have no understanding of the effects of those somewhat *higher* dose-rates we plan to spend considerable resources on to avoid (e.g. by relocation, decontamination, etc.). In other words, we do not know how necessary these dose avoidance strategies are for optimal human health or at what dose-rate it <u>becomes</u> necessary to perform them.

A.3 SOARCA evaluation of health impact follows state-of-the-art approach.

As a society we have developed strategies for dealing with our lack of knowledge of the health effects from low dose-rate radiation. We need these strategies to guide radiation protection policies. For routine radiation protection our limited understanding of the potential hazards presents little difficulty, mostly due to the specifics of this scenario, namely who controls the radiation source, who bears the risk, and who bears the costs of keeping the doses very low. [This is discussed further in Point 5.] Can these radiation protection strategies also be used to project long-term health effects from accidental exposures? Caution is often expressed *against* extending these

strategies to predicting the long term effects of small doses to a large population [2-4], however, as discussed in the SOARCA documentation, few recommendations for precisely *how* to project the effects of small doses have been provided by agencies or committees involved in generating risk estimates. Therefore, in the absence of a better approach, this caution is routinely ignored by the scientific community in situations where the potential magnitude of the impact of low doses is of interest and the result is that the general approach taken in radiation protection is nearly universally employed.

Two health consequences of elevated radionuclide levels in the environment are considered in the SOARCA study: (i) early deaths due to the acute radiation syndrome, and (ii) latent cancer fatalities (occurring many years later). Given the expected rate of progression of the various accident scenarios and the anticipated success of evacuation plans, the risk of acute fatalities (which will only occur following very large radiation doses) is either zero or very, very low. Radiation-related risks then become latent cancer fatalities resulting primarily from exposure to the long-term, chronic radiation dose-rates encountered upon being allowed to return home.

To estimate the risk of latent cancer fatalities from elevated dose-rates in the environment, the dose-rates are first integrated over a 50-year period to derive a total dose. This dose is then multiplied by a risk factor (risk of death per Sv) to determine risk of cancer fatality. Risk factors are from NUREG 6555 [5], and are based on mean responses of 13 experts who provided their estimates of the risk of a latent cancer fatality following a large (1 Gy) whole body radiation dose delivered very quickly (over 60 seconds)[1]. As long as the dose in the first week of the accident scenario is below 0.2 Sv, the doses are assumed to be "low dose rate" and a dose-rate effectiveness factor (DREF) of 2.0 is applied to the risk estimate. In other words, the risk of long-term chronic radiation delivery is assumed to be half of the risk of an acute delivery of the same dose. [More on the use of a DREF in Part B.]

This strategy for determining the risks of exposure to anthropogenic radiation (assuming a threshold for acute effects, integrating the dose over a 50 year period, assuming cancer is the only impact on long-term health, the use of a DREF of 2.0, and the application of a common risk factor throughout the entire dose range) is broadly consistent with the approach taken by several national and international agencies and committees including BEIR, ICRP, NCRP, UNSCEAR, and the EPA [4,6-10]. Thus the approach taken in the SOARCA study for estimating the impact on health of elevated radionuclide levels in the environment has been performed using a state-of-the-art approach.

A.4 Level and depth of new knowledge: reactor systems versus health impact.

It is striking, however, to compare the state-of-the-art related to the impact of low dose-rate radiation on health with the vastly greater depth and detailed understanding we have of many aspects related to nuclear reactors and their subsystems. Significant new information and new analyses have been brought to bear on updated estimates of accident severity since the publication of NUREG/CR 2239 in 1982 [11]. This new knowledge reflects a deeper understanding of the causes and progression of reactor-based accidents through years of development and testing of models of individual systems, and by comparison of theoretical and model-based predictions with measurement data. The huge increase in computational power that has taken place over the same time period has facilitated extensive iterative refinement of the models and, importantly, has made it possible to integrate the models into a comprehensive analysis package in which accident-related changes in one part of the system can be tracked to other parts of the system in a spatially- and temporally-dependent manner.

Thus, while our ability to quantitatively address accident progression has improved dramatically over the last few decades, there has been little change in the depth of understanding of the consequences of radiation exposure to people and we know little more today, about the consequences of living with an elevated dose-rate, than we did 30 years ago.

[1]These risk factors are consistent (within the uncertainty represented by 90% confidence limits and assuming use of a DREF) with those in BEIR V and BEIR VII reports (National Research Council Committee on the Biological Effects of Ionizing Radiation, 1990 and 2005, respectively [6,7]).

This is because we rely on essentially only one dataset (the A-bomb survivor population) to inform our understanding of the long term effects of ionizing radiation on human health. Within that dataset, so few people were exposed to doses relevant to the return-home scenarios addressed by the SOARCA study, that no effect of these radiation doses can be detected with statistical significance, even given the decades-long, high-quality analyses performed on this dataset. Unlike the development of reactor models and accident tracking, which have benefited considerably by orders-of-magnitude improvement in computational power over the past few decades, improved understanding of the consequences of elevated radiation levels on human health has come about only on the time scale of human lifetimes, that is, as more of the A-bomb survivors die and their causes of death are incorporated into our understanding of radiation risk. [See Part B.]

Our limited understanding of the potential consequences of low dose, low dose-rate radiation affects both routine radiation protection scenarios and the accident situation that has led to elevated radionuclides in the environment. In each scenario, however, the implications of our lack of knowledge and the optimal strategies for dealing with it differ considerably.

A.5 Strategies for routine radiation protection are *not appropriate* for use in accident scenarios.

The state of Pennsylvania sets the RHDL at the same dose-rate used to limit dose to the general public from anthropogenic radiation sources in routine radiation protection, 5 mSv (500 mrem) per year. [Information on the setting of dose limits is provided in Part B.] Virginia, which follows EPA guidelines, sets its RHDL a factor of 4 higher for the first year but thereafter matches the 5 mSv/year dose-rate limit used in radiation protection.

When it comes to protecting the public, however, situations involving the unplanned release of radionuclides are *fundamentally different* from those involving routine radiation protection from man-made sources [12,13]. Each situation involves very different trade-offs and these differences should lead to different dose limits. The two situations differ in the level of control over the source of radiation, in the costs associated with keeping doses to the public low, and in who pays these costs.

In the context of routine radiation protection, the source is very tightly controlled [12]. Exposure of the public is allowed to occur but only if the potential risks are smaller than the positive net benefit (e.g. the availability to society of electricity from nuclear power), and even then the risk is kept so low as to be considered trivial (i.e. allowed doses are within the natural fluctuations of background radiation doses [12,13]). Efforts to restrict doses to the public and the financial cost of doing so rest with the owner and producer of the anthropogenic radiation. Any dose-reduction strategies set in place by the owner to protect the public (eg. scrubbers in the stacks), protects all members of the public simultaneously. The owner is actually legally obliged to undertake any 'reasonably achievable' effort to further minimize dose to the public in keeping with the ALARA principle (as low as reasonably achievable). The fact that we do not know how necessary it is, from a health perspective, to keep doses ALARA in the low dose range has become a minor issue, primarily because we are able to keep the doses very low.

This situation is very different from an accident scenario in which radionuclides have been dispersed in the environment. In this case the source of the radiation is no longer controlled. Dose can be avoided, or at least minimized, but only by taking significant and often costly steps. While principles of ALARA can still be applied, the costs (both financial and effort) of applying these principles to avoid or minimize dose have shifted from the source owner to *individual* members of the public as well as to society at large. For instance, while financial reimbursement for some expenses may be available, it is individual members of the public who undergo the upheaval of evacuation, who may need to leave their homes to live in another area (sometimes for long periods of time, perhaps permanently), who face lost opportunity costs, who will be involved in decontamination procedures, who will face prohibitions against consuming local food and water, who may need to abandon farmland or livestock, and who may be urged to spend less time out of doors (since their home will provide some protection against external gamma rays) [1]. Local communities will need to determine what to do with radioactive waste products such as the water from decontamination procedures and surfaces deemed too contaminated to clean, and to make decisions regarding access to such things as community buildings and transportation routes.

With the public now engaging in the efforts for dose avoidance, it is very important that these efforts be clearly justified in terms of the real benefits to their health resulting from undertaking these efforts. At the present time we cannot say that there is a significant impact on health that will be avoided, for instance, by staying away from home, possibly for years [1], until the state-imposed return home dose-rate has been reached. However, we also cannot say that there is *no* impact on health by returning too early. We simply have too little information to address this question.

A.6 The return-home dose-limits in the context of our other radiation doses.

Although we cannot say with certainty what impact the return-home doses will have on health, we can examine these doses in the context of other radiation doses we experience. Figure 1 shows a logarithmic scale of radiation dose on which the average natural background dose to members of the public in the US is indicated (3.1 mSv/year) [14]. This dose comes primarily from isotopes belonging to the ^{238}U and ^{232}Th primordial radionuclide series. Around the world, however, the levels of uranium and thorium vary considerably (by factors of 200 – 400) leading to a large range of natural background radiation doses [15].

Also indicated on Figure 1 are the doses received from a single chest x-ray exam (radiograph) and from a single x-ray Computed Tomography (CT) scan of the abdomen [16]. The use of radiation-based diagnostic medicine has skyrocketed in the last 30 years. In the US we have seen the per capita rate of radiological exams increase by a factor of 10 since the 1980's and nuclear medicine procedures have increased by a factor of 2.5 [17]. Our average per capita dose from diagnostic medicine has increased by about 600% over this time [17].

There were 67 million CT exams performed in the US in 2006 alone; this represents an average of 1 CT exam for every 4 or 5 people in 2006. Some people however, undergo more diagnostic exams than others. Sodickson et al investigated the radiology history of all patients (>31,000) who had undergone diagnostic CT exams at any time during the year 2007 in a tertiary care academic medical center [18]. They found that 33% of all patients who had undergone any CT exam in 2007 had already undergone 5 or more CT exams during their lifetime. Five percent had between 22 and 132 exams and fifteen percent of the 31,000 patients had cumulative radiation doses exceeding 100 mSv. The mean number of exams was 6.1 leading to mean cumulative doses of 54 mSv [18]. While these data reflect the experience in only one hospital, they provide an indication of the doses received by a significant fraction of the population.

For the evacuated public returning home following a severe reactor accident, the doses received during their first year home are also indicated on Figure 1. [The dose to trigger relocation following an accident at the Surry plant used in the SOARCA study (10 mSv) is also shown.] The bases on which the RHDLs are set are not entirely clear. The FDA has suggested use of 2 standard deviations in natural radiation dose as an acceptable radiation risk [19]. In examining "acceptable" risk the EPA compares risks associated with actions already undertaken and accepted by society [1]. However if the acceptability of risk criterion is to be used we must keep in mind that even the least conservative RHDL (20 mSv in the first year) is lower than the natural background doses in many areas in the world. The RHDL for Pennsylvania (5 mSv) is less than a factor of 2 higher than the average background in the US. The dose accumulated from living the first year under RHDL conditions is less than the dose measured from *a single CT exam of the abdomen* (8 mSv) [16].

One rationale the EPA gives for setting the RHDL at 20 mSv is that limiting dose to this level is *reasonably achievable* [1]. It is clear that undertaking the dose avoidance strategies described above will be effective in minimizing dose to the public and thus meeting the dose limit of 20 mSv is achievable. Whether or not it is *reasonable* for the public to undertake these dose avoidance strategies depends on whether they are avoiding a real and significant hazard in doing so. Since the data we use to predict the impact of radiation were all generated at doses and dose-rates much larger than those represented by the RHDL (see Part B), we are ill-equipped to address this question at the present time. Determining the answer to this question should be a high priority; suggestions for proceeding are given in Part C.

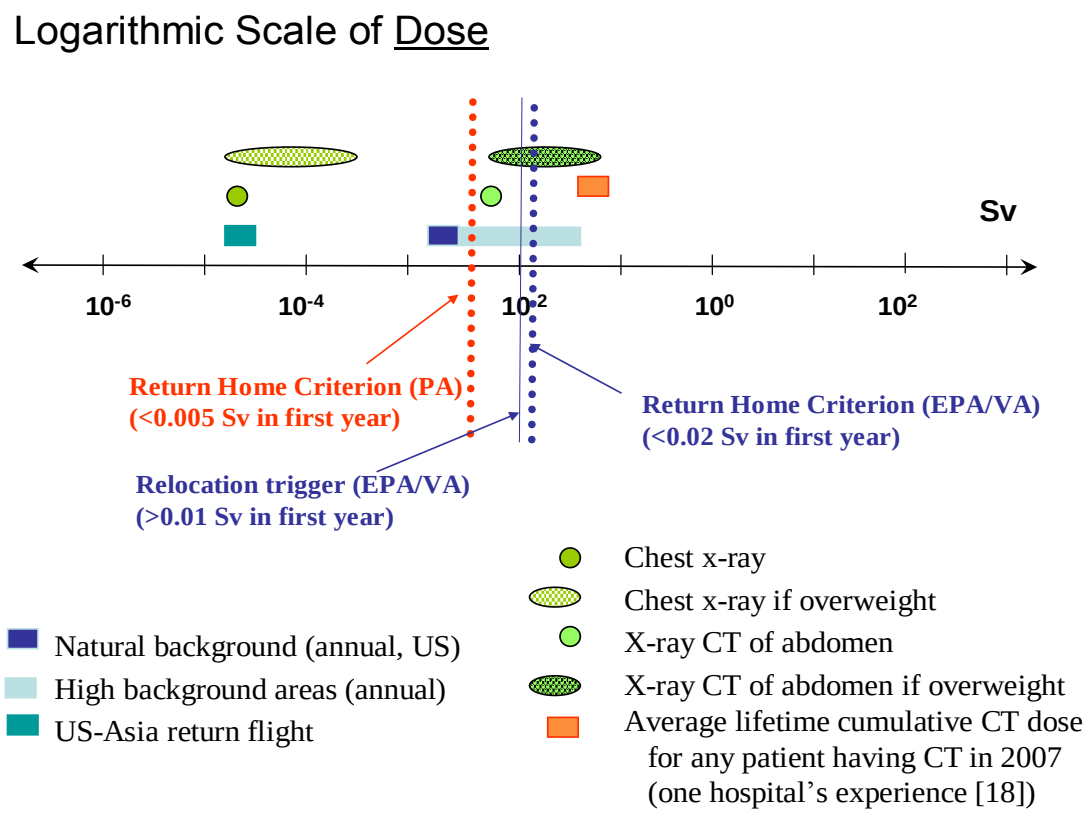

Figure 1. A logarithmic scale of dose showing a range of activities exposing people to ionizing radiation. Shown are annual background dose to residents of the US [14] and to those living in high background regions of the world [15]. Also shown are doses from airline travel and those from radiographic (e.g. chest exam) and CT procedures. [Note that all radiological doses are determined assuming the patient is Reference Man, a thin 70 kg man, 170 cm tall [20]. Since 60% of the population is overweight [21] and since the automatic shut off of the x-ray beam during radiological procedures occurs only when a sufficient number of x-rays has exited the patient, thicker patients require longer irradiation times. For those with only a few cm of extra fat the dose increase is only a factor of 2-5, however since x-ray attenuation increases exponentially with thickness, the dose increase reaches factors of 10 or even more for the very overweight [22]. The average lifetime dose to patients from multiple CT exams [18], shown in orange, is thus an *underestimate*, by an amount that depends on the body fat characteristics (i.e. thickness) of the patients studied.] Vertical lines represent doses used to trigger relocation following an accident at the Surry plant (solid) and those used as return-home criteria (dotted lines).

Part B:
What impact will long-term exposure to low dose, low-dose-rate radiation have on human health?

Throughout the world our estimates of the risks of radiation are based on close examination of the survivors of the A-bomb attacks of 1945[2]. Risk estimates so derived are used to project the long-term effects of *any* exposure to man-made radiation. They are used in the setting of dose limits for occupational exposures or exposure of the general public, for setting Protective Action Guidelines following accidental or intentional (weapons) radiation release, and for setting 'return-home' guidelines, as encountered here.

B.1 The A-bomb Survivors Dataset:

Radiation exposure of the inhabitants of Hiroshima and Nagasaki in August 1945 led to doses ranging from very small to very large, depending on distance from ground zero, but in all cases the dose delivery was very rapid. Because of the height of the blast (1800-1900 ft above the earth), little material from the ground was taken up into the fireball, leading to low levels of radioactive fall-out [23]. Almost all of the radiation dose to the population can be considered as arising from prompt bomb radiation, that is, within the first minute. Long term, low dose rate radiation exposure due to radionuclides in the environment was not experienced.

The effects of large radiation doses were observed within days to months as some of the survivors of the bomb's blast and thermal effects died of symptoms of the acute radiation syndrome. Later, based on results of questionnaires associated with the 1950 Japanese national census [23], 93,700 individuals exposed to A-bomb radiation and 26,600 unexposed persons (residents located more than 18 km from ground zero) were identified and entered into what is now called the Life Span Study (LSS). This study has been ongoing for six decades and is the second longest running epidemiological study of health effects (after the Framingham Heart study) ever conducted.

Individual dose determination for each LSS participant began with a personal interview to determine precise location and body orientation at the time of the blast [24]. Radiation transport calculations then track neutron and gamma fluence from the fireball through various shielding structures between the individual and ground zero. These calculations have improved considerably over the past few decades as a result of greater sophistication in the modeling, newly available interaction cross-section data for important isotopes, finer energy group structure for particle transport, and increased confidence in the dosimetry models resulting from detailed comparisons between model predictions and measured data. Photon fluence estimates at varying distances from ground zero have been compared with thermoluminescence measurements [24] in tiles and bricks that were exposed to gamma-rays from the bomb; neutron fluence calculations have been compared with neutron-induced radioactivity [25,26] in tile, granite, concrete, and soil samples[3]. The ability to compare model-driven fluence estimates with actual measurements taken in various locations and distances from ground zero has resulted in iterative refinement of neutron and photon dose estimates. The doses that individual survivors received are thus now known to a good level of accuracy [24].

Individuals in the LSS are followed until death, and the cause(s) of death recorded. Cancer fatality rates in the exposed and control groups are compared[4] and the relative excess is plotted versus the dose received. Figure 2 is taken from Preston et al (2004), a study that applied the results of the latest dosimetry analysis (DS02) to the cancer death rates observed in the Life Span Study participants; plotted is excess risk of radiation-induced solid cancers (Figs. 2a and 2b) or leukemia (Figs. 2c and 2d) versus dose [27].

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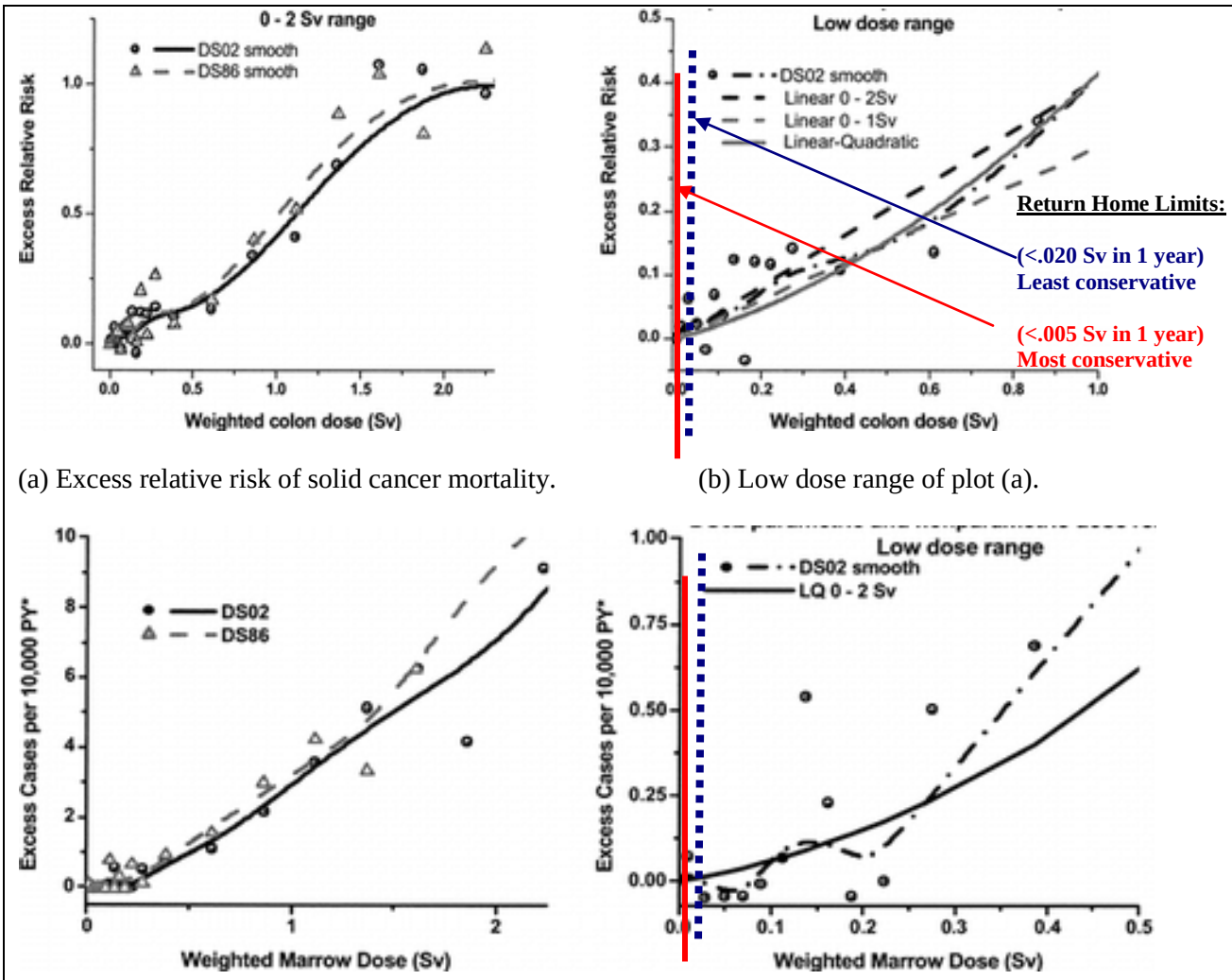
[2] This is the largest and most general population exposed to radiation over a wide range of doses. It is also the largest single cohort of generally healthy individuals exposed to radiation as children.

[3] ^{59}Co (^{60}Co, $T_{1/2}$ = 5.3 yr), ^{151}Eu in tile and granite samples (^{152}Eu, $T_{1/2}$ = 13.5 yr), and ^{35}Cl in concrete (^{36}Cl, $T_{1/2} = 3 \times 10^5$ yr); ^{63}Cu(n,p)^{63}Ni ($T_{1/2}$ 101 yr).

[4] Health effects other than cancer have been examined at high doses; however at low doses non-cancer risks are especially uncertain, according to BEIR VII [7], and are not typically incorporated into risk estimates. Radiation induced mutations in sperm or ova resulting in heritable disease are so low in risk as to be undetectable in humans, even in A-bomb survivors [7].

The dose axes span a very large range (for instance, 2.0 Sv is the equivalent of ~700 times the average natural background dose in the U.S. but delivered within one minute). At large doses (>0.5 Sv) it is clear that survivors are at an increased risk of dying of cancer in later life. At the lowest doses (below ~ 0.1 Sv), the large natural cancer rate in both the exposed and the unexposed populations makes it impossible to declare, with any certainty, what effect small radiation doses have on the cancer fatality rate in exposed persons. But it is precisely this low dose range that we are most interested in. And, since this is the best population available for examining radiation-related health effects, we must use these data in some way to provide guidance for the safe conduct of human activities involving the use or production of ionizing radiation.



(a) Excess relative risk of solid cancer mortality.

(b) Low dose range of plot (a).

(c) Excess absolute risk of leukemia mortality.

(d) Low dose range of plot (c).

Figure 2. Excess cancer risk versus dose. Plots on the left show data points using the prior dosimetry system DS86 (triangle) and the new (DS02) dosimetry calculations (circles). An enlargement of the lower dose data is shown on the right side. Shown are the DS02 data points along with linear fits based on the dose ranges 0 – 1 Sv and 0-2 Sv and a linear quadratic fit based on the 0-2 Sv dose range. [From D.L. Preston et al, "*Effect of Recent Changes in Atomic Bomb Survivor Dosimetry on Cancer Mortality Risk Estimates*," **Radiation Research** (162) 377-89, 2004 [27].]

In 1(b) and (d) the the RHDLs are superimposed on the low-dose cancer risk plots The vertical lines representing the Pennsylvania (5 mSv) and the EPA RHDL (20 mSv) are the doses accumulated over the first year of return-home habitation. [This is an important distinction from the A-bomb doses referred to on the plot's abscissa which were delivered in less than a minute. See more on this point in Part B.3.]

Making Use of the A-bomb Survivor Dataset: To make use of the A-bomb survivor data we must assume some shape of the risk versus dose relationship, then fit an equation to this shape, and use this equation to estimate the risks associated with doses far lower than doses at which statistically significant estimates of risk exist based on the data alone. The equation usually used is a straight line that passes through the origin (0,0). When constrained to pass through zero at the low dose end, the slope of the curve is driven by the risk associated with high doses. With this "linear, no threshold" (LNT) approach, a risk-per-unit-dose estimate is generated (the slope of the line) that can be easily applied throughout the entire dose range. Most agencies and committees involved in evaluating available data and generating risk estimates for ionizing radiation use this approach.

From Risk Estimate to Occupational Dose Limits: This estimate of risk-per-unit-dose is widely used. It is used, for instance, to derive dose limits for workers in occupations dealing with exposure to radiation (after modification for dose-rate effects; see below). Occupational dose limits are set using the "safe industries" argument [28]. The risk to a radiation worker of dying from a job-related cause must be no higher than the risk encountered by workers in safe industries. Using the risk-per-unit dose estimate generated by the LNT approach, the dose that would generate a risk equivalent to that experienced by a worker in a safe industry (on-the-job accidents in retail, government, manufacturing, etc.) is determined and the dose limit is set accordingly.[5] This limit is set at 50 mSv per year.

From Risk Estimate to Public Dose Limits: Dose limits recommended for the worker are reduced by a factor of 10 for application to protection of the public [28]. There are several justifications for setting the limits lower than those for radiation workers. First, the public does not directly benefit (in a wage-related manner) from exposure to the anthropogenic radiation. Second, a wider range of sensitivities is expected to be found in the general population than the adult worker population, and third, the period of exposure can potentially be longer. Dose limits to the public are typically comparable to local variations in natural background radiation [13]. The limit of dose to the public from all anthropogenic radiation sources (excluding medical) is set at 5 mSv per year.

These dose limits (or very similar) to radiation workers and to the public are applied as part of standard radiation protection around the world. They in fact represent minimum standards since every radiation facility is also required to meet the standards of ALARA. Thus radiation workers in nuclear power plants in the US rarely come close to meeting the 50 mSv limit, and the average annual dose to a nuclear power plant worker, steadily declining over the years, is 1.2 mSv (120 mrem) [29]. Similarly, members of the public rarely receive doses at or even near the 5 mSv dose per year limit (with the exception, of course, of their medical doses).

From Public Dose Limit to RHDLs: The EPA recognizes that the recommended upper bound for dose to the public from manmade sources in a single year (5 mSv) was not developed for nuclear incidents and is not appropriate for chronic exposure [1]. They recommend a RHDL of 20 mSv in the first year and 5 mSv thereafter. These RHDLs are shown superimposed on Figures 1b and 1d.

B.2 The accident scenario requires a different approach than for routine radiation protection.

The setting of RHDLs (and other protective action guidelines) at values similar to dose-limits used for routine radiation protection (within factors of several) is problematic for a number of reasons.

First, for decades, all of our risk estimates have been generated for the express purpose of protecting workers and the public from man-made sources of radiation. Because there are few data directly relevant to the impact of low dose-rate radiation, several extrapolations from actual data must be applied. Conservative assumptions, clearly stated, are applied throughout the process. This puts additional onus on the manufacturer/user of the radiation to implement a rigorous system of protection that ensures keeping the dose to workers and the public always very

[5] Note that cancer deaths (including radiologically-induced cancers) occur late in life unlike industry deaths from other causes which lead to an average age of death at 40 years. Thus, radiation leads to fewer years of life lost than with deaths in other industries [28].

low. If the manufacturers cannot maintain these standards of safety they can no longer operate in which case the public continues to be protected. Since doses are, for the most part, very low, the lack of actual data concerning the harm of somewhat higher doses does not become an issue.

However, once elevated radionuclide levels exist in the environment it is not conservative estimates that are needed but actual best estimates of the hazard the public will face if certain dose avoidance efforts are not undertaken. In the post-accident situation, weighty decisions will need to be made regarding: continuing to live in one's home vs. staying away for a long period of time, perhaps forever; allowing vs. prohibiting access to public roads and buildings; selling vs. destroying local produce and livestock [1], etc. The most accurate estimate of the harm generated by the actual radiation exposure conditions is what the public will need at this time.

Second, if we consider the origins of the return-home dose-limits, we have the following steps:

- Using a linear extrapolation of the A-bomb data down to low doses we generate an estimate of the risk of cancer fatality per Sievert.
- We determine the radiation dose needed to result in a risk of death that matches the risk of on-the-job fatality for workers in safe industries such as trade, retail, and government; this dose becomes the maximum permissible dose to workers (50 mSv).
- We then divide the occupational dose limit by 10 to serve as the dose limit to members of the public (5 mSv).
- We then multiply this by 4 (20 mSv first year RHDL, EPA) or not at all (5 mSv per year RHDL, PA) in response to an unexpected or emergency-related release of radionuclides in the environment.

Thus, the public is not permitted to return to their homes until the risk they are exposed to from residual radiation has dropped to 40% or 10% of the risk a government worker faces from an on-the-job fatality. It is likely that most people would consider this level of risk to be minor. On the other hand, permanent or even long-term removal from one's home would be considered a major event. It is unlikely that members of the public would consider this an acceptable trade-off.

Third, the limited data we do have regarding health effects of elevated dose-rate do not support handling routine radiation protection scenarios and those involving long-lived radionuclides in the environment in a similar way. In fact, the data support a different approach for each scenario. The difference comes in assigning the dose-rate effectiveness factor (DREF).

B.3 The Dose Rate Effect:

At average background dose levels in the US, each cell in our body is traversed ("hit") by a secondary electron from natural radiation approximately once each year [30,31]. Around the world, however, there is considerable variation in soil and rock composition leading to dose-rates ranging to more than 100 times the US average [15]. So, the cells of many people around the world experience many more hits per year, as many as 2-3 hits per week.

The doses to the A-bomb survivors were received in less than 1 minute. For those receiving, say, a 2 Sv dose, each cell in their bodies, on average, would have been hit ~700 times, all within 60 seconds. These variations in dose-rate are illustrated on Figure 2 which shows a logarithmic scale of dose-rate in units of both cSv/min and average number of hits/cell per unit time. Superimposed on Figure 2(b) are the return-home dose-rates for Pennsylvania (most conservative) and Virginia/EPA (least conservative). Note the very large difference (at least a factor of 100,000) between the dose-rates encountered by the A-bomb survivors and those we will encounter when returning home, post-accident.

It matters to our bodies how quickly the dose is received. Whether it comes all at once or spread out over a year makes a big difference. This difference is referred to as the "dose-rate effect". This effect has been studied extensively but in general, it has only been studied at those dose-rates that produce an observable biological effect in laboratory studies. The problem is, dose-rates producing a measurable, or even noticeable effect tend to be orders of magnitude greater than those represented by the RHDLs.

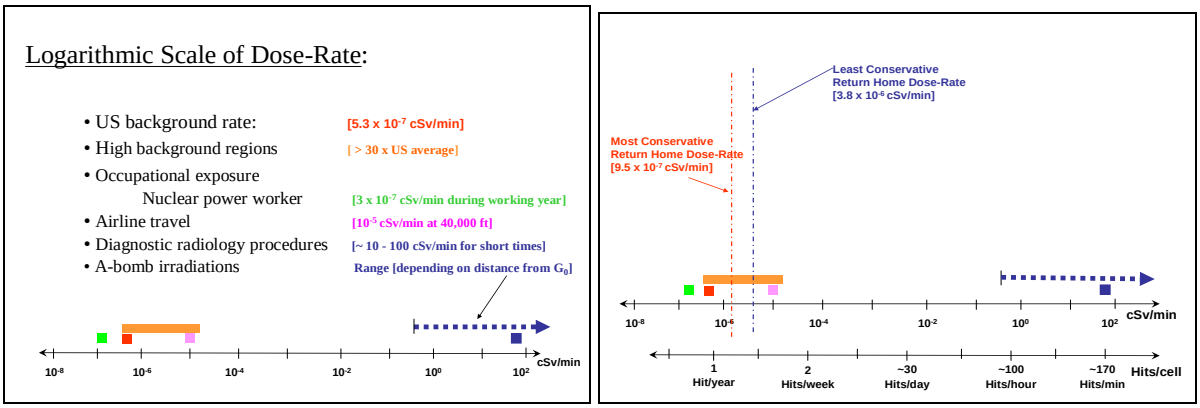


Figure 2. Both (a) and (b) depict the same logarithmic scale of dose-rate. Units are cSv/min (rem/minute). An additional scale is provided on (b) and shows number of radiation tracks (hits) per cell in the body per unit time (assuming 1 hit/cell from an average whole body dose of 3 mSv [30,31]). Shown in (a) are dose-rates from a variety of natural and anthropogenic radiation sources. In (b) the dose-rates corresponding to the return-home dose-rates, averaged over the first year, are superimposed on the data provided in (a).

NCRP 64 "Influence of Dose and its Distribution in Time on Dose-Response Relationships for Low-LET Radiations" (1980) investigated a dose-rate effectiveness factor (DREF) to be used in situations when radiation dose is delivered over time as opposed to delivered acutely [32]. The DREF allows us to take the risk estimates generated using the A-bomb survivor data and apply them to occupational exposures.

In examining the available data, NCRP 64 distinguished between two irradiation scenarios: "protraction" and a "true dose-rate effect". Protraction refers to:

> "exposure times constituting a significant or sizeable fraction of the life span…. Long enough to permit age-dependent changes in the radiosensitivity of the target (e.g. changes in susceptibility to tumor induction or expression with age)" [32].

On the other hand, the true dose rate effect describes shorter-term exposures not influenced by factors important in protraction effects (i.e. includes effects of DNA repair but not of age) [32].

Published data from laboratory studies examining dose-rate effects were examined. Longer term exposures were evaluated separately from shorter-term exposures allowing for separate determination of "protraction factors" (PF) and "dose-rate effectiveness factors" (DREF). Each factor was estimated by fitting high dose-rate and low dose-rate data to linear relationships, both of which were constrained to pass through (0,0), and then taking the ratio of the slopes of these lines.

Clear differences in magnitude were noted between PFs and DREFs. Protraction Factors for cancer induction generated by long-term irradiation ranged from 6.6 to 12.8 with a mean of 10. That is, long-term, low dose-rate radiation is approximately 10 times less effective in cancer induction than the same dose delivered acutely. Shorter-term exposures generated an average DREF of 4 (range 1.1 – 10). The magnitude of the dose-rate factor clearly depends on the total length of the irradiation [32].

In summarizing dose-rate effects for radiation-induced tumorigenesis, NCRP 64 recommends DREF values of 2 – 10. All subsequent examinations of the dose-rate effect, both in later NCRP publications and by other committees, rely heavily on NCRP 64 while also reviewing relevant experimental studies of the dose-rate effect published since that time. A DREF of 2 (or 1.5) is typically applied when establishing low dose-rate risk estimates. There are two reasons for using a low value of DREF. First, most committees have made the assumption that the ratio of impact between high dose and low dose radiation is the same as the ratio of impact between high dose-rate and low dose-rate radiation. That is, the dose effect factor (DEF) will be the same as the dose-rate effect factor (DREF). [Accordingly, these factors are often combined and referred to as the DDREF.] Data from the A-bomb survivors show a reduction by a factor of 2 between risk estimates generated at a dose of >

3 Gy and those generated at a dose of 1 Gy[6]. And second, lower values of the DREF are considered more appropriate for routine radiation protection because they are more conservative. For instance, in choosing to adopt a DREF of 2 rather than a larger number, BEIR V states that the higher values of DREF listed in NCRP 64 reflect situations involving continuous daily irradiation until death, but found that this "may be an unlikely circumstance for humans except as a result of natural background radiation." [2]

Residents returning home, post-accident will, in fact, be irradiated for the remainder of their lives (albeit to an ever decreasing extent) from residual ^{137}Cs in the environment. Therefore the impact of dose protraction rather than merely the true dose-rate effect *should* be taken into consideration when RHDLs and relocation triggers are considered.

As noted in NCRP 64, very long-term radiation reduces the biological consequences to a greater extent than is predicted when just using the DREF [32]. Thus, extending use of a DREF of 2 (or lower) to situations involving very long-term irradiation due to radionuclides in the environment represents a significant overestimate of the detriment. This overestimate, according to the assessment of NCRP 64, is approximately a factor of 5. BEIR VII considers a DREF of 1.5 to be most believable while "…recognizing that the choice is somewhat arbitrary and may be conservative." Overly-conservative risk estimates may have a role to play in routine radiation protection scenarios, but they become unnecessarily burdensome in the post-accident scenario when their use requires important response on the part of each individual member of the public.

Absence of data in the relevant dose-rate range: Also important regarding the analysis of dose-rate effects is the fact that none of the data examined in evaluating the DREF (or the DDREF) have been generated at dose-rates close to those of the RHDLs. Figure 3 reproduces the logarithmic dose-rate plot shown in Figure 2(b) and superimposes the dose-rates compared in each of the studies referenced by NCRP 64 in their summary table from which the DREF for tumorigenesis is obtained. Note that the lowest dose-rate examined in establishing the DREF [32] is still a factor of 100 greater than the least conservative RHDL; most of the "low" dose-rates studied are more than 1000 times greater than the RHDLs. A review of the studies cited by subsequent agencies and committees in discussions of the DREF since NCRP 64 show that this is still the case [4,6-10]: none of the laboratory studies or human exposure conditions examined address the dose-rates encountered upon returning home following a severe reactor accident[7].

One of the reasons lower dose-rate data have not been used is the fact that biological effects at lower dose-rates could not be observed. In fact, NCRP 64 found that low doses and low dose-rates lead to *increased* longevity rather than the decreased lifespan seen at higher doses and dose-rates. In addressing the apparent life lengthening at low dose-rates, the NCRP interprets this effect as reflecting "a favorable response to low grade injury leading to some degree of systemic stimulation." They go on to state that "…there appears to be little doubt that mean life

[6]The basic assumption is that as dose-rate is lowered, a rate is eventually reached (the limiting dose-rate) where the element of time no longer matters and the effects of low dose radiation are the same as low dose-rate radiation. Confirmation that this is so, however, is based on examining dose-rates and doses that are much larger than the RHDLs. For instance, BEIR VII cites only Cornforth et al [33] in support of this assumption. This study found no difference in chromosome aberration number when two 'low' dose-rates were compared (0.105 and 0.047 cSv/min). The lack of difference was interpreted as indicating that both dose-rates must be lower than the limiting dose-rate and therefore no change in biological effect of delivery rate is seen. However, these dose-rates differ by less than a factor of 2.5 from each other, but are more than 10,000 times higher than the least conservative RHDL. It has not been demonstrated that dose-rates consistent with the RHDLs generate the same effect as dose-rates 10,000 times higher when the same total dose is given.

[7] Some data regarding repeated delivery of low doses from diagnostic radiology (sometimes referred to as 'fractionation') are used but, from the perspective of the irradiated cell, these remain high dose-rate delivery scenarios, separated in time. BEIR VII lists six studies involving low dose-rate radiation in their Table 2-1 "Dose-Response Relationships at Relatively Low Doses" [7], however none of these include dose-rates near the RHDLs (with the exception of one study that used Sellafield workers exposed to a lifetime dose of 50 mSv as a *control group* for evaluating the health impact of larger cumulative doses [34]).

span in some animal populations exposed to low level radiation throughout their lifetimes is longer than that of the unirradiated control population." [32, p 104]

Thus, the consequences of radiation exposure at dose-rates used in estimating the DDREF are quite possibly different than the consequences of the much lower dose rates typical of the RHDL. Since no data relevant to the dose-rates of the RHDLs have been evaluated in generating the DDREF, it is not possible to predict with any certainty how to modify the risk estimates generated from the A-bomb survivor population for application to the post-accident return-home scenario. Clearly, however, the uncertainty associated with applying the radiation protection risk estimates to the post-accident, long-term irradiation scenario is very large.

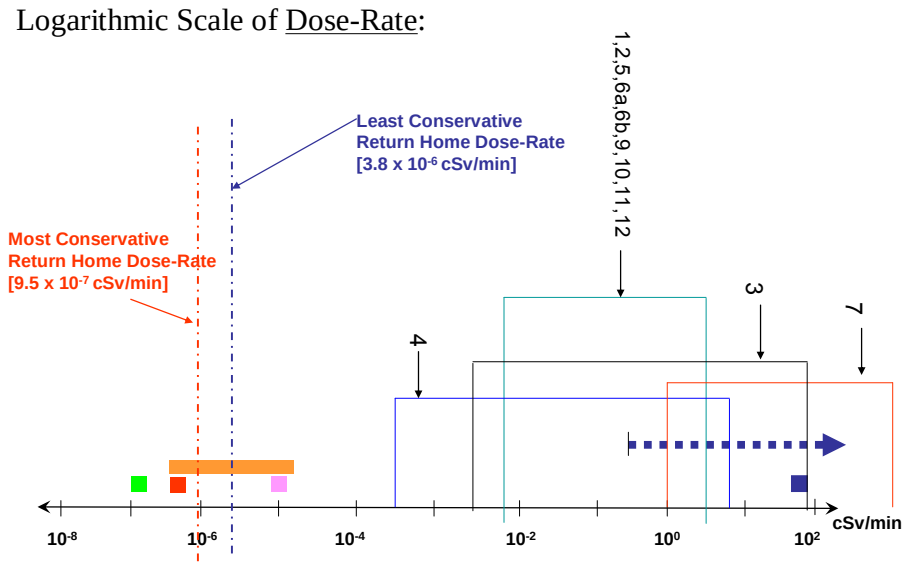


Figure 3. The logarithmic dose-rate chart shown in Figure 2(b) is displayed again here; superimposed are the dose-rate comparisons made by studies referenced by NCRP 64 in estimating dose-rate effectiveness factors (DREF) for tumorigenesis. A total of 12 datasets are cited; the dose-rates compared by each study are indicated, numbered here according to their number in Table 9.3, NCRP 64 [32]. Note that none of the comparisons examined dose rates similar to the RHDLs, or even within a factor of 100.

B.4 Uncertainty in the risk estimates:

As described above, the estimates of radiation risk are very uncertain. Uncertainty analyses of the risk per unit dose estimate generated using the LNT model for application to radiation protection have been performed by a number of committees and agencies [35,36]. Relative uncertainties about the nominal risk estimates are generally estimated to be in the range ± 200 to 400 % [7] when constraining the risk versus dose relationship to pass through (0,0). That is, the 90% confidence interval about the nominal risk estimates covers a range of risk estimates that varies by a factor of approximately 7.

Figure 4a is from NCRP 126 "Uncertainties in Fatal Cancer Risk Estimates used in Radiation Protection" (1997) and shows the ranking of seven contributions to the estimated 200-400 % uncertainty [35]. Interestingly, statistical uncertainties contribute little to the total; this is because the uncertainty estimate is based on the entire dose range including both where uncertainty is small (at high doses) and where it is much larger (at low dose).

The largest contribution to uncertainty (38%) comes from estimating the DDREF. Figure 4b shows the values of DDREF considered in the uncertainty analyses and their relative probabilities of being correct, as subjectively assessed by the NCRP [35] and the EPA [36]. Both agencies assigned a most likely value of 2.0 with the probability of larger values diminishing rapidly. No consideration is made of DREFs as large as the PFs (which are more appropriate to the return home irradiation conditions) but this would *substantially increase the uncertainty* associated with the risk estimates. [Recall the protection factor ranges from 6.8 – 12.8.]

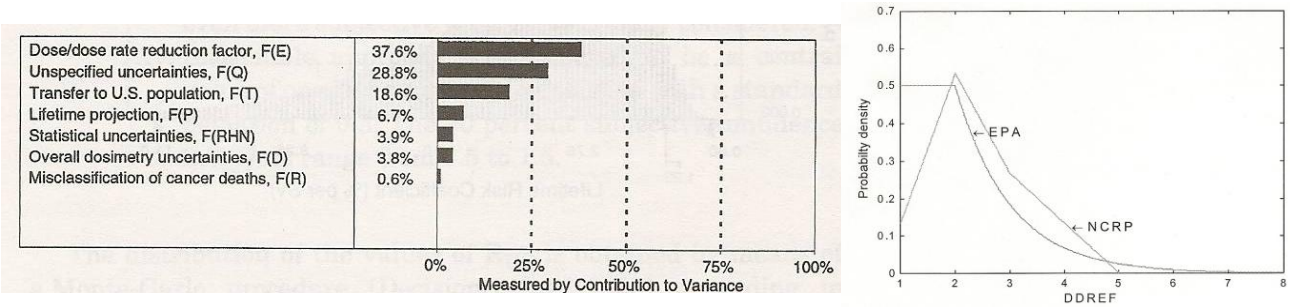


Figure 4. (a) Relative importance of various components of uncertainty to the uncertainty associated with lifetime cancer fatality risk estimates for a general population. Taken from NCRP 126 (p. 72) [35]. (b) Probability distributions for DDREF applied in the uncertainty analyses by the NCRP [35] and the EPA [36]. Taken from [36].

Assuming a very large DDREF (i.e. DDREF → ∞) implies the existence of a threshold dose below which no increase in cancer fatality would be seen. The latest evaluation of the A-bomb survivor data [37] has demonstrated that a threshold dose of 0.04 Sv (4 mSv) fits the data with a statistical significance level *equal* to that observed when using the LNT model[8]. NCRP and EPA clearly state that the 'choice of dose-response model' is not considered in determination of the 200-400% uncertainty in low dose-rate risks estimates (see Figure 4a). However, if both the LNT and a threshold of 0.04 Sv (4 rem), acutely delivered, fit the data equally well, this surely implies an even greater uncertainty in the risk estimates[9].

In summary, the 200-400% uncertainty in risk estimates derived for radiation protection scenarios is likely to be a very significant underestimate of the uncertainty when applying these risk estimates to the effects of long-term, protracted exposure. This estimate of the uncertainty (i) does not consider values of DREF consistent with PFs, (ii) the data used to generate DDREFs do not include dose-rates consistent with the RHDLs, and (iii) the possibility of the dose-response relationship not following the LNT model is not considered. The uncertainty in applying radiation protection based risk estimates to the return-home scenario is likely to be at least a factor of 10 and is probably much greater.

Given this enormous uncertainty, can we really say that it makes sense to require big-impact actions on the part of the public based on doses that differ by factors of only 2 or 4? [For instance, the factor of 2 difference between the maximum allowable dose to members of the public from all man-made radiation sources for routine radiation protection (5 mSv) and the dose levels that trigger relocation of the public following an accident (10 mSv).] It cannot be in the best interests of the public to mandate extensive dose avoidance strategies, when the harm of not avoiding the elevated radiation is so unknown and when the costs for these avoidance strategies are significant and are borne by the public. It is critical that we reduce the uncertainty associated with our estimates of the harm resulting from prolonged exposure to elevated dose levels. Potential strategies for reducing this uncertainty are provided in Part C.

[8] *"Based on fitting a series of models with thresholds at the dose cutpoints in the person-year table, the best estimate of a threshold was 0.04 Gy with an upper 90% confidence bound of about 0.085 Gy. However this model did not fit significantly better than a linear model."* from Preston et al, 2007 [37].

[9] The potential existence of a dose threshold below which no excess cancer fatalities will occur has been considered as part of the SOARCA study by recalculation of the risk data assuming one of three different threshold doses. The use of the threshold models leads to a substantial reduction in the estimated risk of latent cancer fatalities resulting from exposure to radionuclides in the environment.

Part C: Recommendations

The discussion in Part B highlighted the enormous uncertainty we face when trying to predict the impact of chronic, low dose-rate radiation on human health. The SOARCA study demonstrates, however, that it is these dose conditions, almost exclusively, that we will be faced with in the unlikely event of a severe reactor accident.

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.

Decades of intense effort have vastly improved our ability to predict the progression and outcome of a wide range of reactor accidents; predictions have been verified against experiments in numerous subsystems leading to greatly enhanced precision in reactor-based risk estimates. A similar deepening of our understanding has not occurred in the estimation of the impact of prolonged elevated radiation exposure on human health, the most likely consequence of a severe reactor accident. We are getting asymptotically closer to the most accurate risk estimates we'll ever generate with the A-bomb survivor population but, as discussed in Part B, this will not be good enough to tell us what we need to know in the post-accident scenario. We need to examine other irradiated populations and in particular, large populations exposed to low doses, or to low dose-rates.

C1. Diagnostic medical radiation is our largest source of radiation dose: Establish a Medical Radiation Registry for EVERYONE

Some countries operate a registry for tracking annual occupational dose for all radiation workers [38]. 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 [39]), 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.

Initiating and maintaining a database of patient doses, if done correctly, would provide the single greatest database for low dose radiation exposures. It would also present important advantages for risk determination not available with the LSS study. First, it is unlikely the A-bomb survivor data will ever be able to provide a statistically significant determination of the risk of radiation-induced cancer as a function of dose in the low dose range. Stratification of risk estimates to ask questions about radiosensitivities based on age, gender, medical

status, radiation history, etc., will also never be possible in the low dose range with the LSS. On the other hand, hundreds of millions of radiological exams are performed each year in the U.S. on people of all ages. While some radiological procedures are performed on patients with an underlying medical condition that could represent a confounding factor for any future analysis of radiation-induced health consequences, many procedures are performed in cases of acute trauma or in other situations with "no evidence of disease". Even if only a subset of the radiological exams were appropriate for long-term evaluation of the effects of radiation on health we will still quickly accumulate a sufficient sample size for the statistical power we need to answer the question: what impact do low doses of radiation have on our health? With over 350 million diagnostic radiology or nuclear medicine exams performed in the U.S. in 2006 alone [17] the statistical precision possible is very quickly greater than that with the A-bomb survivor study (<98,000 irradiated to any dose in the LSS), and we will be able to stratify the data so that we can assess the impact of low dose radiation on potentially sensitive subgroups within the population.

Second, the population we would be learning about is ourselves. The dataset would not contain the significant uncertainties associated with transporting the risk estimates from a Japanese population (with their substantially different organ-specific baseline cancer rates) to a general US population; this 'risk transfer' is the second largest identified contributor to uncertainty in the risk estimates (see Figure 4a). The Japanese A-bomb survivor population is also different in other ways. The population had been war-ravaged for several years and was malnourished and weakened at the time of the blasts. What impact does this have on our ability to use their cancer risk estimates and apply them to ourselves being irradiated under very different conditions? Multiple stressors on the body make us more susceptible to acute (high dose) radiation effects. Does the same apply to long term cancer induction at low doses? In that case the risk estimates derived from the A-bomb population might be far too high. But what if the effect of the bomb's devastation on societal infrastructure, and access to food, clean water, and medical services immediately after the blast led to the early deaths of the weakest of the population? This would mean that we are now measuring the effects of radiation only on the strong (a 'strong survivor' bias). In that case the risk estimates derived from the A-bomb population study would be too low. Examining the impact of diagnostic medical irradiation on ourselves would eliminate this uncertainty.

There are significant hurdles to overcome in establishing such a database; however it may represent our best opportunity for developing an understanding of low dose effects and for this reason a strong effort should be made to overcome these hurdles. Of great importance to this effort is the current move toward digitization of patient medical records. Over the next couple of years is therefore an opportune time for determining the precise parameters to capture for inclusion in the database and for interfacing with the digital patient records.

For instance, while we do not currently record the doses received by individual patients, strategies do exist for determining individual organ doses for each patient from each procedure[10] and such information could be stored in the patient's electronic record. Other data naturally included in the electronic medical record and of potential relevance to radiation response would be medications at the time of the exam, antidepressants, diet, and prior radiation history.

Analysis of a medical radiation dataset would provide a valuable supplement to the LSS data at low doses, essentially the only dose range of interest in the post-accident scenario. It would not, however, provide direct information about the effects of low dose-rate radiation. [Diagnostic radiology represents high dose-rate delivery, perhaps repeated (e.g. a week or a decade later) to the same or a different part of the body. Given the range of time scales relevant to human biological processes, this situation is unlikely to generate the same biological effect as the same dose spread out continually over time.]

[10] Currently we do not record the doses received by individual patients. Instead we take note of how many radiological exams of a particular type are performed each year in the country, then, on a one-time basis, we measure the dose to a 'typical' patient (using a Reference Man [20] phantom) from this exam. We then multiply the two values together for an estimate of the dose to the entire population, on average, from this particular exam. However strategies exist for determining organ doses for each patient and from each procedure.

C2. Systematic study of the health effects of residents of HBRA around the world:

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 [15,40,41]. Such comparisons, however, often suffer from small sample size, incomplete dosimetry, and a lack of uniformity between studies that prohibits combining of the data. Many comparison studies are ecologic in design in which dosimetry data are aggregated over the entire population; 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 [42,7]. UNSCEAR has suggested that only cohort or case/control studies are suitable for quantification of radiation risk [43].

Similar problems arise with studies evaluating the impact of the Chernobyl accident or large industrial releases of radionuclides. BEIR VII lists a total of 51 ecologic studies related to the health impact following the Chernobyl accident but only 4 case-control studies [7]. Thus, although populations living with elevated radiation levels do exist, we are not yet able to use the information they can provide for the purpose of estimating the impact of elevated, chronic, low dose-rate radiation.

A concerted approach involving co-ordination of efforts across several countries and involving cohort or case-control studies could be undertaken [15,42]. Use of common study protocols and dosimetry methods will improve the robustness of the data and allow data pooling to increase statistical power. A great deal of information relevant to the issue of triggers for relocation and return-home could be obtained from a thorough and long-standing examination of these populations.

Use of such a dataset removes many of the largest sources of uncertainty associated with the risk estimates generated from the A-bomb survivor study (see Figure 4a). First, by directly examining the health impact of large populations living with a range of <u>different dose-rates</u> we can eliminate the largest source of uncertainty associated with using the LSS data for this purpose, namely the error in choosing the DDREF. Second, the uncertainty associated with translating the risk estimates from one population (Japanese) to another (US) will also substantially disappear. HBRA areas are found throughout the world so data will be generated based on similar dose-rates but in different groups with different genetic backgrounds. We will eventually be able to estimate the magnitude of differences between population groups, as well as the magnitude of the uncertainty encountered when applying risk estimates from one group to another. Further, the additional uncertainty associated with "lifetime projection" (see Fig. 4a) will also be removed since with this dataset we would examine actual fatalities, not those that are anticipated.

<u>There are several advantages to such an approach:</u>

1. Dose-rates in HBRA span a considerable range. It would be possible to examine health consequences of a full range of dose-rates including those we are now using to trigger relocation and return-home, as well as dose-rates considerably higher. In addition to examining health of the general population in a HBRA, it would be possible to investigate impact on sub-groups within the population. Analysis of the A-bomb survivor data indicate that women are somewhat more sensitive than men to radiation-induced cancers, and that children are substantially more sensitive than adults with the sensitivity changing remarkably with age at time of exposure [44]. The deeper question is: are the same variations in radiosensitivity expected at low dose-rates and to small doses?
2. The wide range of genetic make-up represented by those living in HBRA around the world provides the opportunity to examine the range of genetic susceptibilities to radiation-induced effects by comparing each group living in an HBRA with its own control group (a similar population but living with lower background levels).
3. There may be particular diets, medications, or even lifestyles that affect radiation sensitivity. For instance, Lemon et al have shown that including a mix of anti-oxidants in the diet of mice results both in increased

longevity and reduced DNA damage (a 6-fold reduction in chromosome aberrations) following a single acute dose of radiation [45]. The neurotransmitter serotonin has been shown to impact the ability of irradiated cells to transmit information about the radiation event to neighboring cells [46]. Would such chemicals as antidepressants, for example, affect the body's response to low dose-rate radiation? And similarly, are there mitigators (medicine) we should be taking when living in an elevated background? Much effort is underway to develop mitigators to redress the effects of high dose/high dose-rate radiation resulting from terrorist activities. Similar efforts could be undertaken to determine if mitigators are useful or necessary when living in slightly elevated backgrounds levels.

4. Given the dearth of data on irradiated humans, animal data are often used to extend our understanding of radiation-induced health effects. However, not all animals show similar responses (as a function of time or of dose) as humans and thus it is difficult to know when and how to extrapolate from animal to human data [47]. Examination of the indigenous animal life, comparing those in the control areas with those in the HBRA, may give us the opportunity to study the magnitude of species-to-species or strain-to-strain differences in response to radiation, perhaps shedding light on the degree to which we can rely on animal data to inform our understanding of human radiobiology.

5. Much of the increase in radiation dose in a HBRA comes externally from elevated gamma-emitting radionuclides in the soil, similar to the irradiation route anticipated with the post-accident return-home scenario. In some areas, however, substantial increases in internal radionuclides contribute to elevated dose through food and water consumption. How different are the effects of higher organ doses when the dose is delivered via external gamma rays than when the elevated dose results from eating food elevated in ^{137}Cs or other radionuclides? In still other HBRAs, the increased radiation is due to higher elevations where contributions to dose from cosmic radiations are increased [40]. These radiations have higher LET but the impact of low dose-rate high LET radiation is not currently well understood.

6. The number of people living in some HBRAs is very large, potentially providing the necessary statistical power to generate statistically significant estimates of the differences in health impact of elevated background radiation. Over 20 million people live in Mexico City (elevation 2240 m) where the cosmic radiation dose is more than double the world average [41]. More than 350,000 residents of the HBRA in Kerala India are currently being studied [42] via interviews to examine factors associated with lifestyle, medications, etc., and dosimetry has been performed in and around over 70,000 homes. Over 125,000 residents of the HBRA in Guangdong Province in China have been under study since 1987 [48].

7. Is an increase in our natural cancer rate the *only* thing we need to worry about? Data arising from analyses of the A-bomb survivors suggest that other health effects may be elevated (although the data are insufficient for detailed investigation). Data from animal studies suggest the relative causes of death as a function of age change after exposure to radiation [49].

C3. Summary:

Both strategies proposed here for developing an improved understanding of the health impact of low dose or low dose-rate radiation will require a very significant effort to generate the data we need to develop our understanding of the real consequences of a severe reactor accident. However there is considerable on-going effort to take advantage of. For instance, experience with life-long tracking of radiation dose is available with radiation registries used to track occupational radiation doses [38], and of course, extensive high quality analysis of long-term health effects from radiation exposure as a function of age and gender has been performed for decades by the Radiation Effects Research Foundation in their analysis of the participants in the Life Span Study [27,37]. This existing experience would represent a valuable starting point, for instance, for a registry of individual patient doses.

The information captured would be of great use to society, and not only for dealing with the consequences of a severe reactor accident. A more precise understanding of low dose radiation effects would be useful in <u>diagnostic medicine</u> (the exponential growth of CT in the US has far outpaced scientific knowledge of the effects of CT-relevant doses on human health; are these doses safe?), <u>space travel</u> (elevated low dose rate exposure, particularly in deep space), and <u>weapons after-effects</u> (far more people will be exposed to low-dose prompt radiation or chronic radiation from fall-out following a nuclear weapons attack than will suffer the acute radiation syndrome).

In all these scenarios we currently rely on the LSS to inform our understanding of the potential health impact of low doses and/or low dose-rate radiation and, as described in Part B, our understanding is very minimal.

For several reasons it is important to begin this effort now. First, the A-bomb survivor study has shown there is a 20-30 year latent period between acute exposure and a cancer diagnosis so it will be many years before we can begin examining health-related consequences of radiation doses. Second, since the move toward digital patient records is underway now, we will lose the opportunity to define which parameters to capture and to store for subsequent analysis of radiation-related impact if we do not use this time to weigh-in on what these records should contain. And third, we need the information that such a database and analysis would provide before an accident occurs. As noted by the EPA, while it would be possible to lower the RHDLs after an accident if it were justified, it would probably not be possible to increase them [1].

As discussed in Part B, most of the data we currently use to evaluate health impact are high dose, high dose-rate data. We need to begin focusing on that dose regime relevant to the post-accident scenario. Emphasis should be on providing best estimate information. One way to get started would be to repeat the Expert Solicitation study (NUREG/CR 6555) [5], asking different questions.

The risk estimates from NUREG 6555 are used, after modification by a DDREF of 2.0, in the quantification of latent cancer fatality risk in the SOARCA study. Questions in NUREG 6555 were put to 13 experts and deal with cancer incidence and cancer death rates for a given radiation dose. For all but one case involving exposure to low LET radiation, the experts were asked to provide their estimate of the risk of fatal cancer resulting from high dose, high dose-rate radiation (1 Gy delivered over 1 minute). Not surprisingly, all experts made use of existing A-bomb-based risk estimates and variation between predictions of the experts was small [5].

Of greater use would be expert estimates of the effects of low dose-rate and low dose gamma radiation, i.e. conditions identical to those we will encounter upon returning home following a reactor accident. [An abundance of recent data relevant to these questions comes from experimental studies of life-long irradiation of small animals, and radiobiological examination of low dose and low dose-rate radiation on cells and tissues in the laboratory (e.g. impact of doses below 5 mGy [50-52]). The field of radiobiology has undergone rapid changes in the last decade with advancements in biological assays and interrogation methods that make it possible to address biological responses at lower doses than possible in decades past.] Expert opinion should be solicited regarding the risks associated with long-term, protracted radiation exposures of the public. Such risk estimates will be far more useful in generating a 'state-of-the-art' estimate of the consequences of elevated radionuclides in the environment.

The industry has done an excellent job of increasing the depth of understanding of reactor technology, accident progression, behavior of fuel and thermal-hydraulic systems under various conditions, radionuclide dispersal, meteorological modeling, particulate deposition patterns as a function of weather, and evacuation planning. Our ability to model and predict the dose that someone will receive from a particular accident initiation sequence that leads to radionuclides in the environment is considerable. However that is where our knowledge stops; we know very little about the impact of this dose (more particularly the dose-rate) on our health. In other words, the real consequence of a reactor accident leading to elevated radiation levels is the impact on human health and we do not know what this impact will be.

We evaluate accident risk in units of "reactor years". As we anticipate increasing our dependence on nuclear power-generated electricity then it is only prudent to develop a thorough understanding of the consequences of the increased radiation exposure we can expect in the unlikely event of a severe reactor accident, an unlikely event whose likelihood increases with every new reactor brought on line or every license renewed.

Appendix

Glossary of Acronyms:

ALARA	As Low as Reasonably Achievable
BEIR	Biological Effects of Ionizing Radiation (US National Academies of Sciences)
DEF	Dose Effectiveness Factor
DDREF	Dose and Dose-Rate Effectiveness Factor
DREF	Dose-Rate Effectiveness Factor
DS02	Dosimetry Study 2002
EPA	Environmental Protection Agency (US)
HBRA	High Background Radiation Area
ICRP	International Commission on Radiological Protection
LET	Linear Energy Transfer
LNT	Linear No Threshold (model of radiation response with dose)
LSS	Life Span Study (A-bomb survivor dataset)
NCRP	National Commission on Radiological Protection (US)
PF	Protraction Factor
RHDL	Return home dose limits
UNSCEAR	United Nations Scientific Committee on the Effects of Atomic Radiation

Units of Dose (and effective dose equivalent):

500 mrem	= .005 Sv = 5 mSv
2 rem	= 0.02 Sv
100 rad	= 1 Gy
100 rem	= 1 Sv
1 Gy	= 1 Sv (for low LET radiation from gamma emitters in the environment)

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Appendix A Peer Review Comments Submitted to the SOARCA Liaison following July 2009 Meeting

Peer Reviewer Comments and Action Items from Kickoff Meeting
July 28 and 29, 2009

Date	Timing of comment/request	Reviewer	Comment or Action Item	Resolution
7/28/09	Schaperow pres., slide 8	Henry	Add common-mode failure to list of items not included in scope. Shutdown and low power also need to be considered to some level of detail since those states have an unknown configuration until the reactor is at full power.	
7/28/09	Schaperow pres., slide 8	Committee	Provide technical justification for each item in the report.	
7/28/09	Schaperow pres., slide 10	Gabor	Defend not including dual plant failures in the report.	
7/28/09	Schaperow pres., slide 11	Committee	Discuss uncertain MELCOR model parameters in the second review meeting.	
7/28/09	Schaperow pres., slide 16	Leaver	Discuss in the document whether "screening" of events is acceptable.	
7/28/09	Schaperow pres., slide 16	Stevenson	Explain in the document why general aviation small aircraft impact is not considered.	
7/28/09	Schaperow pres., slide 23	Leaver, Henry	Consider increased leakage and varying the amount of leakage at different times in the event sequence. Increased leakage early in the accident may lead to higher release. Current PRA may not be adequate. If release into the containment is seen within the first 7-8 hours, SOARCA must be able to field questions about early environmental release. TMI-2 also gives us the perspective that a closed system can release fission products to the containment within a few hours, i.e. when the reactor vessel is intact.	
7/28/09	Schaperow pres., slide 26	Committee	Provide the peer reviewers with table-top exercise mitigation times.	
7/28/09	Schaperow pres., slide 28	Mrowca	In the final report, provide probabilities, or HRA numbers, used for mitigation.	
7/28/09	Schaperow pres., slide 28	Committee	Distribute the HRA report to the peer reviewers, if allowed.	

Date	Timing of comment/request	Reviewer	Comment or Action Item	Resolution
7/28/09	Schaperow pres., slide 28	Stevenson	Consider "aggravated acceleration" by the operators (related to HRA discussion)	
7/28/09	Schaperow pres., slide 28	Stevenson	Consider the use of the term "mitigation". Mitigation implies a reduction of the consequences of an accident or an initiating event. It is also possible that operator or other actions could aggravate accident consequences. The term mitigation appears to bias any action.	
7/28/09	Schaperow pres., slide 28	Mrowca	Add to the report a description of "what is State-of-the-Art about SOARCA?"	
7/28/09	General Discussion	Henry	Significant differences exist between TMI-2 practices and current practices for training and accident analysis which have reduced the potential for radioactive releases to the environment. To the historical perspective in Volume 1, add a section identifying the post-TMI-2 improvements in training and analyzing the spectrum of accident scenarios. Several improvements are listed at the end of this document.	
7/28/09	General Discussion	Committee	Provide the public version of the Executive Summary to the peer reviewers after revisions are complete. Note: Review of this document is beyond the charge of the committee.	
7/28/09	General Discussion	Leaver	In the Executive Summary, emphasize mitigation effects. Consider deleting unmitigated results since these are not best estimate. Emphasize what was learned from mitigation analysis.	
7/28/09	General Discussion	Gabor	Industry heavily focused on PRA quality and methods. Relate SOARCA to existing risk informed regulation.	
7/28/09	General Discussion	Leaver, Clement	Add a faster LOCA for completeness. (note from Vierow - There was discussion that such events are of too low a frequency.) In France, faster sequences are used to study the consequences even though they are of lower frequency and not best estimate.	

Date	Timing of comment/request	Reviewer	Comment or Action Item	Resolution
7/28/09	General Discussion	Gabor	SOARCA needs to have the claim that it has captured all of the risk. Therefore, completeness is needed.	
7/28/09	General Discussion	Stevenson	A Station Blackout may not be the worst consequence of a seismic event. A seismic event in the 10^{-6} to 10^{-7}/yr probability of event range may be sufficient to cause by fault displacement, liquification, or subsidence a movement that could rupture the containment and cause structural collapse or rupture of RCS piping or components. This potential needs to be addressed to show hopefully such events are below the 10^{-7}/yr threshold for consideration.	
7/28/09	Shiekh pres.	Gabor	Provide the peer reviewers with long term drywell temperatures for Peach Bottom scenarios. There is concern about later temperature failures.	
7/28/09	Wagner pres. on Peach Bottom, slide 5	Gabor	Penetration failures should be considered. Without RPV depressurization, instrument tube and CRD tube ejection may dominate and could occur early.	
7/28/09	Wagner pres. on Peach Bottom, slide 14	Henry	If $CsMoO_4$ is modeled, then methyl iodide is also needed. The document reads that $CsMoO_4$ is modeled because it was seen in Phebus. If this is true, then methyl-iodide should also be tracked.	
7/28/09	Wagner pres. on PB, slide 18	Mrowca	The assumption that the diesel generators “fail to start” is questionable. PRA uses “fail to run”, therefore the analysis is conservative.	
7/28/09	Wagner pres. on PB, slide 18	Leaver	Battery life may be another item for a sensitivity study.	
7/28/09	Wagner pres. on PB, slide 18	Henry, Mrowca	Look at the SRV fully open and partially open in the Peach Bottom analysis of long term SBO, i.e. make sure that failure to a fully open state is not used as a significant benefit.	
7/28/09	Wagner pres. on PB, slide 19	Gabor	SRV NOT sticking open should also be considered in sensitivity analysis with impact on potential for penetration ejection as vessel failure mode.	

Date	Timing of comment/request	Reviewer	Comment or Action Item	Resolution
7/28/09	Wagner pres. on PB, slide 23	Henry	Confirm whether separators and dryers remain supported in the Peach Bottom long term SBO.	
7/28/09	Wagner pres. on PB, slide 23	Henry	Consider Te reaction with unoxidized zircaloy (and therefore Te reaction with Sn)	
7/29/09	Wagner pres. on PB	Mrowca	For Loss of Class IV bus, the SPAR has a stuck open SRV, not battery failure. Boundary conditions for this analysis need to be checked.	
7/29/09	Wagner pres. on Surry, slide 17	Henry	Provide identification of uncertainty parameters, range of parameters and their bases. The value of the review may be compromised of the peer reviewers are not made aware of the uncertainties to be considered.	
7/29/09	Wagner pres. on Surry, slide 46	Clement	Unmitigated STSBO with TI-SGTR: The hot leg failure occurs 15 minutes after SGTR, therefore most FP's go into containment. An uncertainty study can be done on preventing hot leg failure and waiting for a pressure vessel failure. (Some reviewers agree, however SNL noted that the analysis does not approach a high pressure vessel failure.)	
7/29/09	Wagner pres. on Surry, slide 19	Committee	Provide the peer reviewers with Dana Powers' memo on ARTIST DF's for SG tubes.	
7/29/09	Wagner pres. on Surry, slide 33	Mrowca	Unmitigated short term SBO: There is the concern that if these procedures are published in a NUREG, the licensees may want to take credit for them.	
7/29/09	Wagner pres. on Surry, slide 40	Mrowca	Mitigated short term SBO: the water supply needs to be confirmed. Procedures must exist for injecting water.	
7/29/09	Wagner pres. on Surry, slide 41	Gabor, Henry	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?	

Date	Timing of comment/request	Reviewer	Comment or Action Item	Resolution
7/29/09	Wagner pres. on Surry	Stevenson	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.	
7/29/09	Wagner pres. on Surry	Committee	Consider the state of the steam generator tubes in the Surry analysis.	
7/29/09	Bixler pres., slide 7	Clement	5 rem/yr is now 2 rem/yr.	
7/29/09	Bixler pres., slide 9	O'Kula	Ensure text is consistent with meteorological data provided. Discuss how a "representative year" is chosen from data that varies widely, or how a sensitivity study will be performed to confirm year in question is appropriate. For example, p. 58 of Vol. I shows different predominant wind direction for Peach Bottom (2005 and 2006) and large precipitation difference for Surry (2001 and 2004).	
7/29/09	Bixler pres., slide 9	Yanch	Explain why the RBE for bone marrow is reduced to 1.	
7/29/09	Bixler pres., slide 9	O'Kula	Consider dose conversion factors for children and adolescents for those cohorts that are largely composed largely of those population groups, e.g. "schools".	
7/29/09	Bixler pres., slide 10	O'Kula	Three different references are cited for deposition velocity, are they one and the same? Ref. 48 in Vol. I, Fred Harper et al., NUREG/CR-6244, and USNRC/CEC expert elicitation	
7/29/09	Bixler pres., slide 10	O'Kula	Please provide the draft report of the NRC's interpretation of CEC study, "Expert data report for deposition and relocation", or other bases for deposition velocity.	

Date	Timing of comment/request	Reviewer	Comment or Action Item	Resolution
7/29/09	Bixler pres., slides 12 and 20	O'Kula	The report should indicate what is included and excluded in population dose. For example, food ingestion, decontamination workers, people returning to their homes. Explain from MACCS2 inputs/assumptions, and results, the key parameters affecting population dose.	
7/29/09	Bixler pres., slide 12	Mrowca	Discuss in the report the basis for SOARCA values and mention values used by others, esp. NUREG-1150, for relocation, habitability, etc.	
7/29/09	Bixler pres., slide 16	O'Kula	How do these times for MACCS2 compare with those used for MELCOR? For example, does t=0 mean the same in each?	
7/29/09	Bixler pres., slide 21	O'Kula	Show how health risk impacts can be reduced to various countermeasure criteria (long-term dose) for a given sequence. Possibly tie operating procedures and accident mitigation procedures with early phase risk metrics.	
7/29/09	Bixler pres., slide 33	Gabor	Highlight qualitatively the differences between SOARCA and SST1 results and the general reasons for the differences.	
7/29/09	Jones pres., slide 24 and Bixler's slide 33	Leaver	The timings listing in the slides should be consistent.	
7/29/09	Wagner pres. on Surry, slide 74	Leaver, Gabor	The ISLOCA sequence does not need to be reported. The sequence is not possible because B.5.b equipment would be used. The best estimate is that this sequence won't happen. Gabor: May be true for PB and Surry, but B.5.b is not completely implemented in other plants.	
7/29/09	Wagner pres. on Surry, slide 74	Clement	Mechanical resuspension needs to be addressed if turbulent deposition is to be taken into account.	
7/29/09	Wagner pres. on Surry, slide 74	Leaver	ISLOCA: Once the flow is going, Reynolds numbers will be very large. Turbulent deposition is significant. DF's must be looked at.	
7/29/09	Bixler pres. on Surry, slide 52	Leaver	ISLOCA: Do we want to show calculations out to 100 miles? Will this result in undue concern?	

Comments regarding the post-TMI-2 improvements in training and analyzing the spectrum of accident scenarios

August 4, 2009 email transmittal from Bob Henry to Karen Vierow

My comments regarding the post-TMI-2 improvements in training and analyzing the spectrum of accident scenarios is given below.

The current description of NRC sponsored studies includes the major improvements in understanding and analyzing the responses of representative BWR and PWR designs. These include the Reactor Safety Study (WASH-1400), NUREG-1150 and now SOARCA. In addition to the improvements in understanding and calculational capabilities, there have been numerous influential changes in the training of operating personnel and the increased utilization of plant specific capabilities. For example:

- The transition from event based to symptom based Emergency Operating Procedures (EOPs) for the BWR and PWR designs.
- The performance and maintenance of plant specific PRAs that cover the spectrum of accident scenarios.
- The implementation of plant specific, full scope control room simulators to train operators.
- An industry wide technical basis, owners group specific guidance and plant specific implementation of the Severe Accident Management Guidelines (SAMGs).
- Improved phenomenological understanding of influential processes such as (a) in-vessel steam explosions, (b) Mark I liner attack, (c) dominant chemical forms for fission products, (d) Direct Containment Heating, (e) hot leg creep rupture, (f) Reactor Pressure Vessel (RPV) failure and (g) Molten Core Concrete Interactions (MCCI).
- Proceduralized use of plant specific B.5.b systems.

All of these have contributed to reductions in the likelihood of a severe accident as well as a reduced potential for radioactive releases to the environment. As such, they should also be identified in the historical background for SOARCA.

Comments on SOARCA Report[1]

David Leaver

August 5, 2009.

1. Between the slides and the report it appears that there are five event types which SOARCA does not address: multi-unit events, spent fuel pool accidents, low power or shutdown events, security-related events, and the very large seismic event causing simultaneous breach of containment and a LOCA with ECCS failure. Discussion of the reasons for not addressing these event types is spread out in the report and is somewhat uneven (e.g., for the very large seismic event, extensive discussion is given in the Executive Summary, and multi-unit events are discussed in both Volumes III and IV; security-related events, low power/shutdown events, and spent fuel pool events did not seem to get as much discussion or at least I could not locate it.)

   It is suggested that the reasons for not addressing these five event types be discussed in a more even-handed, consolidated manner, probably in Volume I. The reasons for not addressing a given event type might include, for example: plans exist to address it in the future, it is judged to be low priority, or it is already adequately addressed somewhere else. This discussion is part of the matter of completeness which, along with the screening approach and sensitivities, is very important to the credibility of the SOARCA effort. It is certainly acceptable to carry out the project without claiming to be complete, but the SOARCA effort should be as complete as practical and should deliberately defend its degree of completeness.

2. It would seem appropriate and desirable to benchmark MELCOR fission product releases against the TMI-2 accident and SFD.
3. There was mention of an ongoing HRA study that would quantify the likelihood of success of b.5.b mitigation actions. Will this be complete in time to support SOARCA? Can we see it? See also comment 5.
4. Some comments on sequence screening:
   a. Some of the support points for screening are marginal. For example, the first full paragraph on Vol. I, page xi, justifies 1E-6 as 1% of CDF and uses the 1E-4 QHO as the CDF. But these days, CDFs for U.S. plants are more like 1E-5 to 1E-6, and 1% of this is a factor of 10 or more less than 1E-6.
   b. Another example is in the next paragraph where it is stated, "Another way to judge the impact of low-frequency events is to consider the increase in the latent cancer consequences that would be necessary to offset the lower frequency." This is a good argument and should be used. But what about early fatality consequences which are more visible and will start to show up as frequencies get lower?

[1] It might not be a bad idea to organize the comments into General (comments on higher level issues such as methodology and presentation) and Specific (technical matters, editorial type comments). I have not tried to do that here but if this is thought to be a good idea I will do it

c. It might be wise to cite screening precedents. See, for example, NUREG-1420 which indicates that consequences with frequencies lower than about 10^{-7} per year "...are not meaningful for decision making," and Regulatory Guide 1.174 and the U.S. Reactor Oversight Program significance determination process, among others, which use a frequency threshold for non-risk-significant changes.

d. The best screen is one where you defend its reasonableness and its application, but then show you don't really need to lean on it too much. See comment 5 and comment 20 on the Exec. Summary for one way to do this.

5. For all of the sequence types, the mitigated sequences appear to be the only ones that survive the screen. Using Surry as an example, consider the table below. While we don't know the precise likelihood of success of the mitigation actions, given the time available to the operators to take these actions and the fact that these actions are thought out and planned for in advance with equipment, procedures, and training, we should be able to quantify the likelihood to at least an order of magnitude. For purposes of this comment and in the absence of the HRA study (see comment 3), the success probability of the mitigation action(s) is assumed to be 0.9 except for LTSBO where the additional time available would tend to support a lower number, assumed to be 0.99. Under these assumptions, the shaded sequences are the ones that survive the screen and would represent the realistic, best-estimate characterization of severe accident consequences for Surry. Then, for completeness and to develop insights on the importance of mitigation actions, a series of sensitivity sequences are analyzed which include the unmitigated sequences along with other uncertainties and sensitivities (which are TBD). It may make sense to lump the unmitigated sequences, along with uncertainty and sensitivity results, into something called sensitivity studies rather than call them out separately.

Surry Sequences	Mitigated			Unmitigated		
	Frequency (1/yr)	**Rel. Mag. (I, 48 hr)**	**Release Onset (hr)**	**Frequency (1/yr)**	**Rel. Mag. (I, 48 hr)**	**Release Onset (hr)**
LTSBO seismic	1.98E-5	N/A	N/A	2E-7	0.003 (72 hr)	45
STSBO seismic	1.8E-6	0.006	67	2E-7	0.006	26
STSBO with induced SGTR	4.5E-7	0.005	3.6	5E-8	0.009	3.6
ISLOCA	2.7E-8	N/A	N/A	3E-9	0.095	9.2
Spontaneous SGTR	4.5E-7	N/A	N/A	5E-8	N/A	N/A

6. It is a good idea to do a sensitivity study on later HL creep rupture, but note the point that induced SGTR will hasten the time of HL creep rupture so as to at least qualitatively make the case that significant delay in HL creep rupture after SGTR is very unlikely.
7. Why not include SG injection as a mitigation action for STSBO? Doing this will cut the induced SGTR contribution to I release (currently 0.5%) in half, and will be even more important if HL creep rupture is delayed per comment 6.
8. Turbulent deposition should be considered for the ISLOCA. For Surry, the ECCS pipe length L and pipe diameter D were 80.2 m (263 feet) and 0.1397 m (5.5 inches), respectively. To put this in perspective, the pipe is almost as long as a football field, but its cross-section area is barely that of two fists. Therefore, this is a typical long pipe problem with a large length to diameter ratio, which tends to produce high decontamination factor for aerosols. In Surry the ECCS line has an orifice which results in high Re number flows (1E5 to 1E6 during the time of fission product release). This in turn results in high DFs (range of 10 to 50). The LACE tests also support a DF in this range. Even if only modest DF effect is considered (factor of 2 or 3), this is important for the sensitivity sequences. While the gas flow velocity in the ECCS line is high enough to support turbulent deposition, it is subsonic (no shock wave) which should help the mechanical resuspension issue.
9. The non-fission product to fission product (inert) aerosol mass ratios used for SOARCA modeling seem low based on our work, particularly for BWRs. For PWR-type fuel bundles measurements from the SFD 1-4 experiment indicate inert aerosol mass (Cs, Sn, Cd, Ag, U, others) in the range of 1 to 3 x the fission product aerosol mass. There is also information available from Phebus FP tests which suggests even larger ratios. BWR cores of the same power level as a PWR core have 2 to 4 x the mass of materials that form inert aerosols in a severe accident, and only about 25% more fission product mass. We typically use 1:1 for PWRs and 2:1 for BWRs in our design basis calculations.
10. The bottom paragraph on page 7, Vol. I is not very clear. An example would help.
11. In Figure 20, the containment airborne aerosol reduction at the time of HL creep rupture is very fast. It looks like reduction of a factor of 3 in minutes. We have not seen deposition rates from natural processes (sedimentation, diffusiophoresis, and thermophoresis) this high.
12. The matter of potential radiation exposure to the operator for each of the mitigation actions should be addressed.
13. Vol IV, page 105, second paragraph, 6th line: Should it be “from the vessel”?
14. It is very reasonable to limit dose results to 10 miles as was done in the Executive Summary, based on the NRC safety goal policy. The dose results elsewhere in the report should be limited to 50 miles. There are several good reasons for this: (1) for the interested reader it provides a significant increase in distance beyond the 10 mile results in the ES; (2) the value-impact methodology for backfit is out to 50 miles; (3) the emergency planning ingestion exposure pathway zone is 50 miles; (4) looking at LCF results from the SOARCA reports, there is little change in LCF risk beyond 50 miles (see, for example, Vol. IV, Figure 144, which shows LCF risk for unmitigated STSBO as decreasing by a factor of almost 10 between 10 and 50 miles, but less than a factor of 2 between 50 and 100 miles.); and (5) showing results to 100 miles risks unnecessarily conveying a notion that reactor accidents threaten people out to that distance and beyond.
15. References should be available and traceable (e.g., “Keith Eckerman [51]” should be a memorandum or some such document so the public can access it).
16. Vol. II, page 70, last sentence of first paragraph, and a number of other places, use the term “physically unreasonable” to describe why early containment failure phenomena are no longer considered. This term does not connote the situation very well to me. I would suggest

alternative wording, for example: "While the phenomena are conceivable, the conditions necessary for them to occur in an LWR severe accident environment are so remote that the phenomena are now considered essentially impossible in this environment."

17. SOARCA should include the 0.5% who choose not to evacuate since this is real and is a best estimate. Note, however, if the approach of comment 5 is used, the best estimate has zero early fatalities so this does not affect the best-estimate.
18. A basis should be provided for assuming safety systems and structures (including containment leak rate) function as designed after an earthquake which is 3 or 4 x the SSE. This is also an appropriate matter for a sensitivity study (i.e., increased containment leakage early).
19. Some comments on the Section 6 discussion on Emergency Response. Using Vol. 3 on Surry as an example:
    a. The notion of emergency response out to 20 miles was very prominent in Section 6 and as presented conveys the wrong idea. I suggest toning down the amount of information on 20 mile effort (other than consideration of shadow evacuation which is a realistic consideration of the 10 mile evacuation) and when it is discussed make clear that it is just a sensitivity study.
    b. The references apparently are misnumbered. Also two different ways are used in referring to references (see for example the first paragraph on page 176 ("[10]" and the last paragraph on page 177 ("(NRC, 2005)").
    c. First paragraph on page 179: "WINMACCS allocates 0.061 percent…" should be 6.1 percent.
    d. Really hard to read or figure out Figure 130.
    e. Hard for me to discern Table 18 though if I spent more time maybe I'd get it.
    f. First full paragraph on page 185: "EAL SS1.1 specifies that if all offsite AC power is lost for greater than 15 minutes an SAE is declared" should be all onsite and offsite AC power. This phrase occurs in many other places.
    g. "Cohort 4: 10 to 20 Public" paragraph on page 186: "This was established at 3 hours after gap release." I think this should be at 6 hours after gap release.
    h. Similar comment as f. applies to Section 6.4.1.2 on page 187, i.e., gap release for unmitigated STSBO occurs at 3 hours, not 9 hours.
20. These are placeholder comments on the Executive Summary (ES). Sensitivity and uncertainty results are necessary to finalize these comments and the ES.
    a. The ES should be changed to make more visible the main objectives and conclusions from SOARCA. The objectives are clear and are summarized on slide 4 of the presentation, "SOARCA – Scenario Selection and Mitigation Measures". A text version of these objectives appears in the ES (page ix), but the objectives are somewhat run together and not very visible. Conclusions are given on slide 9 of the same presentation and appear in text form to some degree in the ES but are not succinct and visible.
    b. There should be further discussion on what the important results and conclusions are involving the full peer review group and after sensitivity and uncertainty results are available. It is suggested that the results and conclusions be divided into main, high-level conclusions, and supporting results. Here is a strawman set of main conclusions from SOARCA:
        i. SOARCA represents a major change from the way that the public perceives severe accidents and their likelihood and consequences.
        ii. Severe accident likelihood and consequences are significantly lower than indicated by previous reactor risk studies.

iii. Public health effects from severe accidents are very small. SOARCA results indicate that latent cancer fatality risk from severe reactor accidents is more than a million times smaller than the U.S. average risk of cancer fatalities, and that prompt fatality risk from severe accidents is essentially zero.

c. The main conclusions should be followed by a set of more specific results which support and amplify the conclusions (e.g., accident scenarios progress more slowly with smaller releases; accident mitigation is likely (due to time and redundancy) and would be effective when implemented; emergency response is likely to be effective in significantly reducing health risk)

d. An important result is that the long-term portion of the LCF risk (which is ~90% of the total risk) is controllable. This should be stated in Volumes III and IV and reflected in the ES.

e. The ES should be written around and emphasize the realistic, best-estimate consequence results (i.e., the mitigated sequences). The sensitivity results can then be presented and discussed (including unmitigated sequences, uncertainty results, and other sensitivities). An important point here is that the main conclusions from SOARCA (whatever those end up being – see comment 20 b) apply even when sensitivity results are taken into account.

Appendix B Peer Review Comments Submitted to the SOARCA Liaison following September 2009 Meeting

Peer Reviewer Comments and Action Items from Sept. 15-16, 2009 Meeting

Date	Timing of comment/request	Reviewer	Comment or Action Item	Resolution
9/15/09	Schaperow pres.	Canavan	As an EPRI project, Surry is updating their seismic PRA. The complete PRA is expected to be completed in early 2010. Canavan will inquire as to whether he can share preliminary results.	
9/15/09	Schaperow pres.	Stevenson	Foundation failure has not been addressed sufficiently. It has been analyzed at Savannah River, as an example. Foundation failure needs to be addressed as a generic failure, not plant specific. (Schaperow noted that this was left out since evaluation capabilities are not currently sufficient.)	
9/15/09	Schaperow pres.	Henry	Consider whether catastrophic containment failure should be addressed. (Schaperow noted that the probability is about 10^{-7}, which is below the criteria of 10^{-6} unless it is a bypass. This was left out since evaluation capabilities are not currently sufficient.)	
9/15/09	Burns pres.	Canavan	NUREG-1855 (EPRI 101 6737) reports on treatment of uncertainties in risk-informed applications. The SOARCA team should refer to this report. (Leonard noted that epistemic portions will apply.)	
9/15/09	Wagner pres., slide 5	Vierow	The probability of a thermally induced SGTR was noted to be just above the screening criteria. The assumption of a stuck-open SG safety valve at 3 hours may reduce the sequence probability below the screening criteria. This is a good example of an event retained for completeness. Include Tinkler's explanation in the final documentation that other analyses consider safety valve leakage to obtain the high pressure differential-low SG water level conditions.	
9/15/09	Wagner pres., slide 14	Gabor	Is a Decontamination Factor of 7 still valid late in time when flow rates are reduced?	

Date	Timing of comment/request	Reviewer	Comment or Action Item	Resolution
9/15/09	Wagner pres., slide 19	Henry	The assumption of "no UO_2 present after vessel failure" needs to be justified. There may be some reactor designs in which not all of the debris exits the core region. Some Westinghouse designs have upflow and downflow (KV - in the downcomer?) which allows a fraction of the debris to remain. (Wagner said that they may need to consider Ru release. He noted that a ring of fuel may remain in the lower plenum.)	
9/15/09	Wagner pres., slide 21	O'Kula	The graph on Slide 21 shows unreferenced data, and was said to be from a draft NUREG with Dana Powers as the lead. Please provide a reference for the graph on this slide.	
9/15/09	Wagner pres., slide 26	Stevenson	Detonation needs to be examined, not just deflagration. There is a factor of 3 difference in pressure.	
9/15/09	Wagner pres., slide 26	Canavan	Canavan will provide data to Schaperow on spray patterns at low flow rates (less than 2/3 rated flow) for containment sprays. This data should be reflected in analysis.	
9/15/09	Wagner pres., slide 28	Leaver	Consider whether it is possible to have a single burn that could lead to detonation.	
9/15/09	Leonard pres., slide 5	Mrowca	Provide SPAR models for Peach Bottom and Surry, if possible.	
9/15/09	Leonard pres., slide 9	Henry	Add implications of steel failure, both static and dynamic.	
9/15/09	Leonard pres., slide 12	Leaver	How do we know that the valves will function after sitting open and exposed to hot fluid?	
9/15/09	Open discussion	Henry	The definitions of "sensitivity" and "uncertainty" are needed. These will promote the decisions as to which sequences and cases need to be analyzed. For example, with the thermally-induced SGTR, does the base case quantify risk?	
9/15/09	Open discussion	Henry	An approach to quantify or bound movement of structures in the BWR is needed.	

Date	Timing of comment/request	Reviewer	Comment or Action Item	Resolution
9/15/09	Open discussion	Henry	Buoyancy flows in the containment are not part of the calculations. They need to be discussed, along with the concern that any cases that are more important are not being neglected.	
9/15/09	Open discussion	Leaver	The matter of completeness may be the most critical issue we have. How can the story on completeness be made? The Executive Summary was unevenhanded regarding completeness. (Schaperow noted that SOARCA is a truncated risk study.) • How does the NRC make the case for completeness? • For events just below the cutoff frequency, how can their deletion be justified?	
9/15/09	Open discussion	Gabor	We have a base method for performing consequence analysis, as has been presented to us. How do we incorporate results of sensitivity calculations into the consequence analysis?	
9/15/09	Open discussion	Mrowca	The connectivity between thermal hydraulic consequences and risk is weak.	
9/15/09	Open discussion	Yanch	There may be more completeness than is stated in Volume 1 of the draft NUREG. The case needs to be made better. Add more references and point to more data. There is too much assuming what the reader already knows.	
9/15/09	Open discussion	Leaver	Elaborate more on the screening process in the document.	
9/15/09	Open discussion	Mrowca	Consider relooking Level I. State-of-the-Art was not done for seismic or fire PRA. It was used at the end of the analyses.	
9/15/09	Open discussion	Leaver	A systematic discussion that screened sequences are not fundamentally different from the ones looked at is needed.	
9/15/09	Open discussion	Gabor	LERF represents about 10% of the core damage frequency (CDF) by industry data for PWRs. This is inconsistent with SOARCA and will need to be explained.	

Date	Timing of comment/request	Reviewer	Comment or Action Item	Resolution
9/16/09	Jones 1st pres., slide 6	Kowieski	Why is siren used as particular points? It gives the impression that people move at this time. Suggest changing to "siren + ES message".	
9/16/09	Jones 1st pres., slide 6	Kowieski	Reconsider the 1 hour allowed to evacuate after second siren. (SOARCA team requested feedback from the committee on this 1-hour time.)	
9/16/09	Jones 1st pres., slide 6	Vierow	Sensitivity studies could be done here. Some parameters are plant specific, e.g. bus availability, while others are random, e.g., weather, time of day. These should be distinguished in the report.	
9/16/09	Jones 1st pres., slide 10	Kowieski	The evacuation time of the Special Facilities is late and will not go over well with the public.	
9/16/09	Jones 1st pres., slide 6	Canavan	Specify when each group is notified in order to show that none of them are being neglected.	
9/16/09	Jones 1st pres.	Gabor	Is a loss of ac power a unique event? It may lead down a path that is different than for a non-blackout event. Blackout may not be conservative. Consider when EAL is triggered.	(Same as later question in Open Discussion.)
9/16/09	Jones 1st pres.	Leaver	The effect on risk of the declaration of EAL (Emergency Action Level) needs to be captured.	
9/16/09	Open discussion on Emergency Planning	Yanch	The public session should be opened with a statement on where SOARCA is conservative. This will give the public a better understanding of the thought processes and methodologies behind the analyses.	
9/16/09	Open discussion on Emergency Planning	Leaver	Assess the sensitivity on the time to declare a General Emergency (GE). Even if the sensitivity is low, that is valuable information.	
9/16/09	Open discussion on Emergency Planning	Leaver	Measure the sensitivity of health effects to the speed of declaring a GE. For example, a LOCA does not survive the screening process but could it have health effects?	
9/16/09	Open discussion	Canavan	The conclusions need to be documented better throughout the NUREG. Too much is left for the reader to interpret.	

Date	Timing of comment/request	Reviewer	Comment or Action Item	Resolution
9/16/09	Open discussion	Gabor	With the Station Blackout conditions for the long term (transient), use different EALs and see effects. Try normal EALs, not the SBO EALs.	
9/16/09	Bixler 1st pres., slide 5	Leaver	Discuss the best way to present the data. Consider showing a histogram to see the differentials.	
9/16/09	Bixler 1st pres., slide 5	O'Kula	The y-axis will be confusing to the public. It is a conditional risk, or risk given that the accident (STSBO) has occurred. So risk here is not per year, but per the accident occurring. If we say "risk" alone, it should factor in the mean estimate of the frequency (3E-07) and show units on the order of 10^{-11}. We will need to have these plots be standardized one way if "conditional risk" results are portrayed, and another way if absolute risk is being shown. As it stands now someone will see the y-axis numbers and misinterpret the result, e.g. try to relate it to meeting the safety goals.	
9/16/09	Bixler 1st pres., slide 6	Stevenson	Note that "mean" is conservative with respect to the "median".	
9/16/09	Bixler 1st pres., slide 5	Leaver	The data is extremely important but may lead to a negative perspective. Consider deleting this data in the NUREG.	
9/16/09	Bixler 1st pres., slide 16	Kowieski	Too much time is spent on the non-evacuating public.	
9/16/09	Bixler 1st pres., slide 16	Leaver	The evaluations can be done on the basis of 100% evacuation, therefore the early fatality risk is zero.	
9/16/09	Bixler 1st pres., slide 18	Leaver, Kowieski	There is a strong precedent for presenting only out to 50 miles of data. Consider not showing the 100-mile data.	
9/16/09	Bixler 1st pres.	Canavan	Make comparisons to voluntary or involuntary exposure to assist the public with understanding the doses.	
9/16/09	Bixler 1st pres., slide 20	Gabor	Eliminate the original results in the report and show only the latest cases with the new cohorts.	

Date	Timing of comment/request	Reviewer	Comment or Action Item	Resolution
9/16/09	Bixler 2nd pres., slide 4	Yanch	Calculate for different weather conditions as a sensitivity study. It is important to report the consequences of bounding weather conditions, along with the consequences of mean weather conditions.	
9/16/09	Bixler 2nd pres., slide 4	Canavan	Pick a specific rainy day and a specific sunny day, since these days really happened, and analyze under these conditions. This can be used to justify the mean.	
9/16/09	Open discussion	Leaver	Applying the LNT seems inconsistent with the habitability criterion. (note from KV – I am not sure I have this recorded comment correctly.)	
9/16/09	Open discussion	Leaver	Land contamination and security events are missing from this report. The security events, in particular, may likely draw claims of missing events.	
9/16/09	Open discussion	Stevenson, Leaver	The concern remains about increased leakage due to seismic events. The concern is particularly for PWRs. An expert is needed to help define the fragility of leakage. A possible reference is the SQUG (Seismic Quality Uncertainty ???) data on fragility.	
9/16/09	Open discussion	Leaver	The completeness argument is fundamental. • Address the fact that there are no cliffs lurking below the screening cutoff • If security arguments are not to be addressed, state that security events are not expected to have an effect on SOARCA results. • With respect the Human Reliability (HRA), mitigation actions are considered in the SOARCA and they could drive the sequence below the screening cutoff.	
9/16/09	Open discussion	Yanch	Some data is referred to as coming from the utilities. Consider adding an independent source so that there is not an appearance of having flavored data.	

Date	Timing of comment/request	Reviewer	Comment or Action Item	Resolution
9/16/09	Open discussion	Gabor	• For the completeness story, focus should be on the Level I selection and screening process. • H_2 burning sensitivity – a delay in hydrogen burn should be analyzed (at higher H_2 concentration) • Calculate the BWR Main Steam Line heatup without assuming a stuck open SRV. In addition, run a case without the SRV failing open, but with a Main Steam Line failure.	
9/16/09	Open discussion	O'Kula	The MELMACCS treatment of source terms needs to be better explained. As discussed in the draft Vol. I and plant-specific Vols. III and IV, there is a wide gap in the discussion from once the source term is determined to the point where the evacuation, sheltering, and normal activities are modeled. There needs to be more discussion on how the MELMACCS mode transitions the MELCOR output to forming WinMACCS input, the assumptions applied, etc.	
9/16/09	Open discussion	O'Kula	In Volume I, add lessons learned since NUREG-1150, and what is leading to the reduction in risk for these selected sequences. Are we smarter with our methods and tools? Have experiments given us insights that we didn't have before? Have any of the post-TMI requirements improved the outcome? Is it better operating training that eliminates sequences? What is driving the reduction acute and latent risk? If Volume I is the most read of the SOARCA NUREGs, then let's be clear on the sources of reduction in risk. {If the final report from NUREG-1150 is read, you get an appreciation on the changes between WASH-1400 (1975) and NUREG-1150 (1990)}.	

Peer Reviewer Comments and Action Items
By Roger B. Kowieski – 10/02/2009

Date	Timing of comment/request	Reviewer	Comment or Action Item	Resolution
Present., 9/16/09	J. Jones pres., Slide 8	Kowieski	Slide 8 (Peach Bottom) suggests that after declaration of GE by the plant, sirens and EAS message could be activated within 45 minutes. Based on the actual field experience, it takes approximately 15 minutes for the nuclear power plant to notify the state authorities, and may take additional 38-40 minutes, before the sirens activation and EAS message are completed. Therefore, total time required to complete the A/N sequence may vary between 53-55 minutes.	
Present., 9/16/09	J. Jones pres., Slide 16	Kowieski	Slide 16 (Surry) suggests that after declaration of GE by the plant, sirens and EAS message could be activated within 45 minutes. Based on the actual field experience, it could take up to 60 minutes to complete the A/N sequence (Sirens/EAS message).	
Present., 9/16/09	J. Jones pres., General observation	Kowieski	It appears that the existing documents do not address the notification of public in case of siren(s) failure. Should a siren fail, it may take additional 45 minutes to notify the affected public by Route Alerting procedures.	

Date	Timing of comment/request	Reviewer	Comment or Action Item	Resolution

Additional Comments on SOARCA Report

David Leaver

October 5, 2009.

1. So as to make the frequency cutoff more robust and less of a black and white process, it would be prudent to examine an order of magnitude or so below the frequency cutoff to confirm that there are no sequences with consequences that might significantly exceed those already being considered in SOARCA or that might impact overall conclusions which are derived from the best-estimate, baseline sequences. To an extent, SOARCA has already done this by virtue of including Surry interfacing LOCA which came in at less than 10^{-7}, including Peach Bottom unmitigated STSBO which is less than 10^{-6}, including Peach Bottom Loss of Vital AC Bus E-12 which was less than 10^{-6}, and including the unmitigated sequences which when quantified even in a conservative manner should drop below the cutoff. But it needs to be documented and presented in the report as part of, or a backup to, the screening process.
2. Volume III, Section 3.1.4.1 is confusing. It states that, "One unmitigated case was considered." But then it goes on to discuss two unmitigated cases: a first case with RCIC black run and use of portable power supply credited, and a second case with RCIC black run and portable power supply not credited.
3. Supplement to August 5, 2009, Comment 1: Of the event types that were not addressed in the draft report, the most important is security events, particularly airplane crash. A study such as SOARCA will lose credibility and impact if it is silent on this. It is recognized that for confidentiality reasons, there is limited information that can be presented on security events; plus it may only be possible to characterize probability in a qualitative manner. But there is much that could be said about what the Commission has done to address these events, and the limited consequences which are expected (e.g., no more significant than the sequences that are analyzed explicitly in SOARCA).
4. Delete August 5, 2009, Comment 17 and replace with the following: Regarding the matter of the 0.5% who choose not to evacuate, it is suggested that results be reported for non-voluntary risk (i.e., 100% evacuation) and that the voluntary risk (for those who choose not to evacuate) be reported as part of the sensitivity study.
5. A summary of fragilities for key components (e.g., Surry low pressure injection and containment spray; PB torus integrity, RCIC) for the 0.3 to 1 pga earthquakes would be useful, or at least the basis for assuming that they can perform their function after the earthquake. Both Surry and Peach Bottom are members of the Seismic Qualification Users Group (SQUG) which was developed by industry for older plants and may have some useful data. Dr. Robert Kassawara (650 855 2775) is the EPRI Program Manager for SQUG. NRC is aware of the SQUG database, having considered it in conjunction with resolution of USI A-46. NRC's Goutam Bagchi was involved in this. The EPRI seismic margins report (NP 6041, Rev. 1 – a licensable document) may also be useful.

6. The LCF consequence curves (such as Volume III, Figure 64 and Volume IV, Figure 145) might be more meaningful if the risk was presented for a given radius (or ring of some average radius) as opposed to plotting the risk to all residents inside a given radius.
7. SOARCA indicated that it is pursuing this, but just for the record, the Ba release for Peach Bottom STSBO both without (Figure 38) and with (Figure 45) RCIC Blackstart looks very suspicious. It is 4 x the iodine release early, and ends up nearly the same as iodine in the longer term, in the range of 6% to 8%.
8. The table below is an attempt to show the Peach Bottom sequences that were analyzed. The following comments apply:
    a. There are no mitigated STSBO sequences (i.e., no STSBO sequences with 10CFR50.54(hh) measures considered). What is the reason for this? Apparently Peach Bottom had not yet procured the required portable equipment as of the time of the site visit, yet the 10CFR50.54(hh) portable pump is credited in the Peach Bottom mitigated LTSBO (see Volume III, Table 4). For STSBO without RCIC blackstart, RPV pressure is less than 100 psi after about 4 hours, and lower head failure does not occur until about 8 hours. For STSBO with RCIC blackstart, these times are even longer. It would appear that there is time to put the portable pump in place to achieve a benefit, possibly preventing lower head failure, or at least delaying lower head failure, and also reducing radionuclide release.
    b. 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.
    c. If the Peach Bottom mitigated STSBO sequences are considered, the unmitigated STSBO sequences would then become sensitivities, and should be retained in the spirit of comment 1 above on looking below the frequency cutoff.
    d. The Loss of Vital AC Bus E-12 sensitivity for operator failure to manually depressurize and failure to open CRDHS throttle valve has core damage, but there is no radioactive release analysis.
    e. If the sensitivity for Loss of Vital AC Bus E-12 with operator failure to manually depressurize and failure to open CRDHS throttle valve is included, a probability should be estimated. The frequency would likely be an order of magnitude or more below the <1E-6 number that is given in the report for the base case.

PB Sequences	Mitigated			Unmitigated		
	Frequency (1/yr)	**Rel. Mag. (I, 48 hr)**	**Release Onset (hr)**	**Frequency (1/yr)**	**Rel. Mag. (I, 48 hr)**	**Release Onset (hr)**
LTSBO Seismic	1E-6 – 5E-6	N/A	N/A	1E-7 – 5E-7	0.037	19.5

STSBO, no injection Seismic	???	???	???	1E-7 – 5E-7	0.1	8
STSBO, RCIC blackstart Seismic	???	???	???	1E-7 - 5E-7	0.075	13.5
Loss of Vital AC Bus E-12	<1E-6	N/A	N/A	??	??	??

9. In Volumes III and IV, Sections 6 (EP) and 7 (Consequences), it appears that the unmitigated sequences are given undue emphasis. For Volume III (Peach Bottom), per Table 9 all 3 of the scenarios assessed for emergency response are unmitigated. For Volume IV (Surry), per Table 15 4 out of the 5 scenarios assessed for emergency response are unmitigated. Emergency response and consequence analysis of unmitigated sequences is appropriate as a sensitivity, but why not have a best-estimate, base case which uses sequences that survive the screen? Based on the August 5, 2009 Comment 5 table, there are two such Surry sequences with a non-zero release (mitigated STSBO and mitigated STSBO with induced SGTR). There may not be any non-zero release sequences for Peach Bottom that survive the screen, but the next closest sequence could be considered (either the unmitigated LTSBO or the mitigated STSBO) for the base case so as to have a Peach Bottom release for the best-estimate, base case consequence and emergency response analysis.
10. Land contamination results probably do not belong in the SOARCA reports, but was there any condemned land in any of the sequences?
11. Volume III, page 8 – Second full paragraph: "The process identified two sequence groups which met the screening criteria of $1x10^{-6}$ per reactor-year for containment failure events…" looks wrong. Should it not be "…$1x10^{-6}$ per reactor-year for core damage frequency"?
12. Suggested parameters for uncertainty and sensitivity analyses:
    a. Higher confidence weather. The risk from this (i.e., the higher LCF consequences together with the lower frequency of the higher confidence weather) can then be compared with the risk from the mean weather.
    b. Habitability criterion (e.g., cut by a factor of 5, and/or vary the costs used in the decision as to whether contaminated areas can be restored to habitability). See Volume I, page 65 and 67.
    c. Relocation criteria (e.g., what is additional LCF risk for 5 rem for normal relocation?) See Volume I, page 66.
    d. How about a no ad-hoc evacuation sensitivity case?
    e. Time for mitigation measures (e.g., 8 hours for transporting and connecting the Surry diesel-driven injection pump could be increased to 12 hours). See Volume I, page 23.
    f. Aerosol deposition velocity in consequence calculations. See Volume I, page 64.
    g. Shielding factors. See Volume I, page 65.

h. Time of Declaration of GE. See, for example, Volume IV, Figures 131 and 132, which have GE at 2 hours. The paragraph above Figure 131 says, "It is assumed under this scenario, that plant operators would recognize rather soon that restoration of power within 4 hours is unlikely. A 2 hour period from loss of power was selected as a reasonable time for declaration of a GE…" This certainly is reasonable, but the plant operators could also think that power might be restored and thus delay the declaration of GE a bit longer, say until 3 hours.
i. Delay times for shelter and evacuation – See Slide 7 of the September 16, 2009 EP presentation. By inspection, modest differences in the delay times won't matter much, but it is good to demonstrate it.
j. What is the effect of degradation of containment leaktightness due to an earthquake in the 0.3 to 0.5 pga range, and in the 0.5 to 1.0 pga range? For example, consider DBA leakage x3 for 0.3 to 0.5 pga, and x10 for 0.5 to 1.0 pga.
k. This matter was brought up in one of the first two meetings by Jeff Gabor. What about a sensitivity on the radionuclide release assuming that the SRV sticks closed after excessive cycles (see Volume III, Figure 31)?

October 6, 2009

(From John Stevenson)

Dear Karen:

Please find herein my suggested corrections to my second comment on page 6.

"The concern remains about increased leakage due to seismic events. The concern is particularly for PWRs. An expert is needed to help define the potential for leakage as a function of cracking in concrete."

As a result of the review of the SOARCA report and discussions held at the two-review group meetings, I believe there are two areas which may require further attention.

- Seismic Issue

In general, at the mean 10^{-4}/yr probability of exceedence frequency level effectively used for the design of existing U.S. NPP[1] seismic induced foundation failures are not credible. These failures could in general be from three causes.

  - Failure of reactor building foundation due to earthquake fault motions propagating to the ground surface beneath the reactor building,
  - Failure of saturated cohesionless soil including engineered backfill reactor building foundations by liquification during the earthquake or settlement due to consolidation following the earthquake.
  - Failure of buried piping that potentially leads to containment penetration failure.

While such foundation failures as described above are not considered credible at the 10^{-4}/yr earthquake hazard probability level, typically applicable to NPP design, I am not aware of any studies that have made at the 10^{-6}/yr earthquake hazard level that has been defined as the threshold event frequency for this study. Reactor building foundation material is either rock or engineered backfill. Foundation failures have been observed for commercial structures on unimproved foundational materials as shown in Attachment A, and its potential occurrence at a NPP site because of improved foundation materials should be below the 10^{-6}/yr probability level.

It is my expectation that fault movement surface propagation under the reactor building is not credible event at the 10^{-6}/yr earthquake hazard level due to the explicit evaluation of capable faulting during initial NPP siting, but it is not clear that this would also be true for liquification and settlement phenomenon at the earthquake 10^{-6}/yr hazard at that level. Most U.S. NPP sites at the 10^{-4}/yr hazard level have mean peak ground accelerations, pga that would be in the range between 0.2 and 0.3g.

The slope of seismic hazard curves typically are between 2 to 3 times the pga for a factor of 10 decrease in frequency in the range of 10^{-4} to 10^{-6}/yr. This suggests that pga's for a 10^{-6}/yr earthquake probability would be between 1.0 and 2.0g. Beside acceleration level it is also important for liquification or

[1] For existing U.S. NPP seismic hazards were determined deterministically and were subsequently evaluated probabilistically where seismic SSE loads were determined to be between 10^{-3} to 10^{-5}/yr.

settlement to consider strong motion ground shaking duration which might increase from 20 to 30 seconds to more than 1.0 minute.

In summary, it may very well be the case at the 10^{-6}/yr mean threshold level that earthquake induced foundation failures of engineered backfill cohesionless saturated soils will not be credible for reactor building foundation and penetration failure or containment and/or RCS foundation failure, but I do not believe this potential has been sufficiently evaluated to date.

- Hydrogen

The potential for hydrogen deflagration within containment as a result of a LOCA appears to have been carefully studied particularly with respect to steam inerting which precludes hydrogen reaction with oxygen. However, there does not appear to have been a distinction made between hydrogen deflagration (burning) which may occur several times without steam inerting during the course of LOCA with hydrogen volume percentages below 10 percent and detonation (explosion) of hydrogen concentrations above 10%. Existing containment design can be expected to accommodate hydrogen deflagration without failure, but the potential for a hydrogen detonation with a resultant pressure load at or near the containment failure load should be evaluated explicitly.

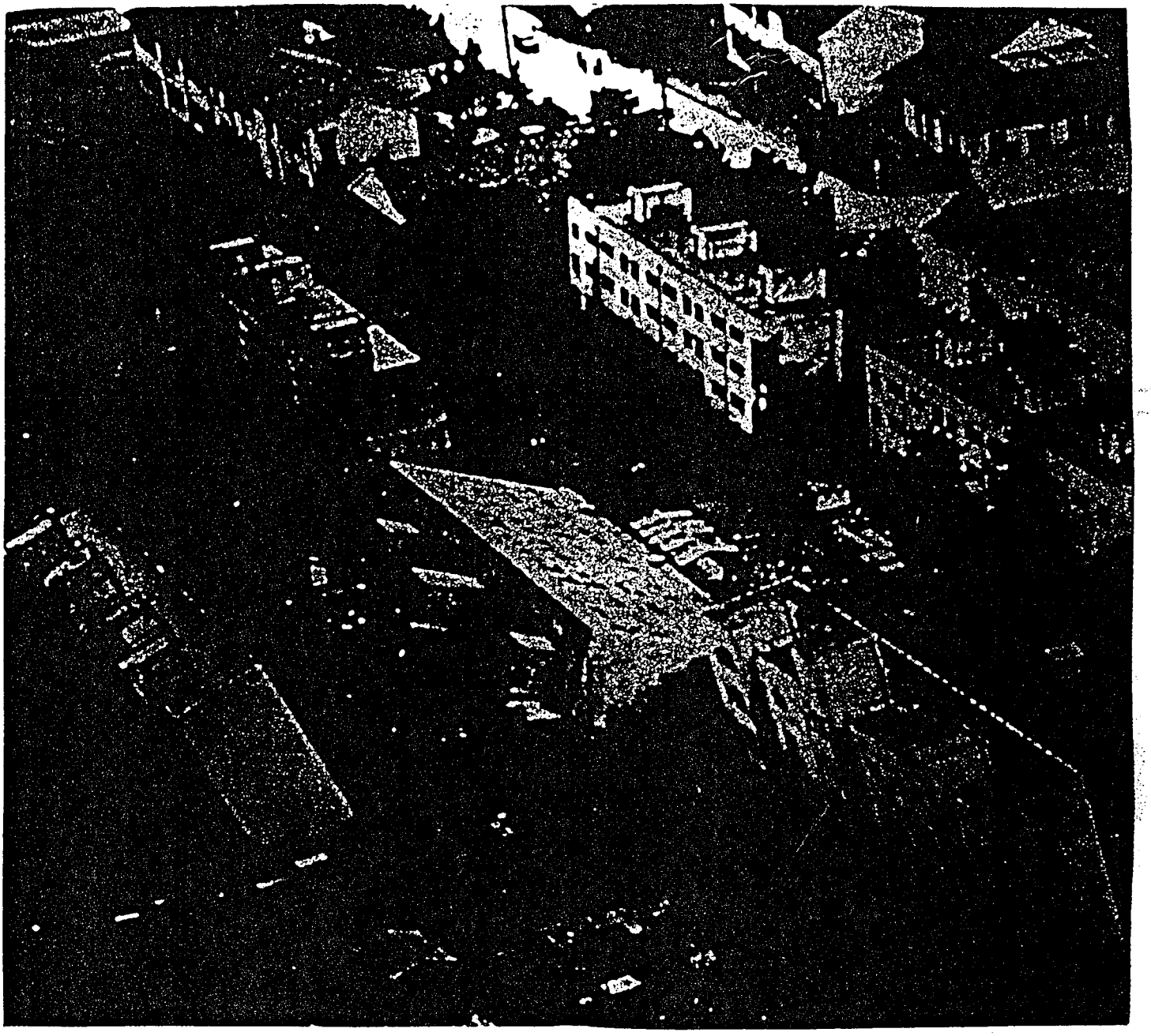

FIGURE 9 TILTING OF APARTMENT BUILDINGS AT KAWAGISHI-CHO DUE TO SOIL LIQUEFACTION RESULTING FROM THE NIIGATA EARTHQUAKE, JUNE 16, 1964.

Comments/Questions on SOARCA Volume I

COMMENT AND RESOLUTION SHEET						
Document Title: State-of-the-Art Reactor Consequence Analysis (SOARCA) Project SOARCA Methods Volume I				**Doc. No. NUREG-XXXX/SAND2008P-XXXX**	**Doc. Date:** July 2009	**Date Comment Sent:** 21 August 2009
Commenting Individual or Organization: Kevin O'Kula, WSMS				**Phone**/Email: 803.502.9620/kevin.okula@wsms.com	**Resolution by:**	**Phone No**.
No.	**Page**	**Section**	**Type: M – Major Med - Medium Min –Minor E -- Editorial**	**Comment, Question**	**Resolution of Comment**	
1	ix	Background and Objective	Med.	NUREG/CR – 2239 and NUREG/CR- 2723 are both cited as being referenced throughout the SOARCA documentation. I have been able to download the latter (Strip report) but the former (Sandia Siting Study) does not appear to be available on the web. Can this be made available to the review panel?		
2	xi		Editorial	2nd paragraph, 2nd line: American Society of Mechanical Engineers'		
3	3	1.0 Introduction	Editorial	A introductory, transition sentence or two is needed ahead of the first paragraph on page 3. The paragraph reads as though it is the present tense, e.g. " Yet the possibility remains . . .". Suggest a statement to note that it is in reference to the state of knowledge during or after WASH-1250.		
4	15	Section 2.2	Minor	Suggest that first use of SPAR models be noted with a citation/reference.		
5.	22	3.11	Minor	Was short-term Station Blackout from a seismic event for Peach Bottom included or dropped?		
6	57	5.1	Medium	Is the selection of METCOD still based on machine time considerations? Would runs using METCOD=5 be too machine-intensive to run? Is there a technical basis for LHS more so than Stratified Random Sampling (METCOD=5; with NSMPLS=24; so that every hour of the 8760 hour data set is sampled)?		

Comments/Questions on SOARCA Volume I

COMMENT AND RESOLUTION SHEET						
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Commenting Individual or Organization: Kevin O'Kula, WSMS				**Phone**/Email: 803.502.9620/kevin.okula@wsms.com	**Resolution by:**	**Phone No**.
No.	**Page**	**Section**	**Type: M – Major Med - Medium Min –Minor E -- Editorial**	**Comment, Question**	**Resolution of Comment**	
7	58	5.2.1	Medium	Table 12 shows characteristics of the two years of meteorology considered for each plant. For Peach Bottom, the predominant wind changed by nearly 180 degrees (SSE to N). For Surry, the number of hours with precipitation went from 388 to 521. Was any work done to determine why one year was more representative over another year in each case?	The question only pertains to Surry. The windrose figure answered my question for Peach Bottom.	
8	64	5.4	Medium	Deposition velocity is an area where the uncertainty analysis capability in WinMACCS could offer a big improvement over the point value selection process that was applied in previous studies. It would be of interest to use the uncertainty capability in the new suite of MACCS2 modules to see the impact of the parameter values used in the 9-, or 10-group deposition velocity distribution.		
9	**64**	**5.4**	**Minor**	**Similar to 8 above, how would different values for the surface roughness length change the risk results at the mean (average) level? Could a short paragraph or limited sensitivity analysis be used to address whether this is important within the 10-mile EPZ, and within the 20-mile region?**		
10	Throughout	5	Major	What kind of larger uncertainty analysis for the overall SOARCA project is envisioned? Will there be any attempt to examine aleatory and epistemic classes of uncertainties?		
11	**64**	**5.4**	**Medium**	**Ref. 48 (Bixler, N.E., *Expert Data Report*, Sandia National Laboratories: Albuquerque, NM) is described as the expert elicitation study for deposition velocity. Could this report be made available to inform the review panel of the values used? If it's the same as Harper, F. T., et al., "Probabilistic Accident Consequence Uncertainty Analysis, Dispersion and Deposition Uncertainty Analysis," NUREG/CR-6244, 1994, it is no longer needed.**	**Now have the reference.**	

Comments/Questions on SOARCA Volume I

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Document Title: State-of-the-Art Reactor Consequence Analysis (SOARCA) Project SOARCA Methods Volume I				**Doc. No. NUREG-XXXX/SAND2008P-XXXX**	**Doc. Date:** July 2009	**Date Comment Sent:** 21 August 2009
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12	**64**	**5.4**	**Med**	**The fourth paragraph states:** **Aerosol deposition velocities are calculated by MELMACCS based on the geometric mean diameter of each aerosol bin, as defined in the MELCOR analysis. The deposition velocities are based on expert elicitation data using the median value of the combined distribution from the experts [48]. Typical values for surface roughness and mean wind speed, 0.1 m and 2.2 m/s, respectively, are additional parameters used to determine the deposition velocities in MELMACCS. Mean wind speeds were determined from the specific weather files used in the consequence analyses.** **MELMACCS is being relied upon to perform post-processing of MELCOR results to provide a set of deposition velocities for MACCS2. To understand this set of inputs, and the basis for their preparation, we would need to see a discussion/document on MELMACCS to describe its technical basis, and the inputs used to generate the sets of deposition velocities. In addition, a table is needed, if not in Volume I, then in Volume III (Peach Bottom) and Volume IV (Surry), on the input deposition velocities used for the MACCS2 analysis.**	

Comments/Questions on SOARCA Volume III

COMMENT AND RESOLUTION SHEET						
Document Title: State-of-the-Art Reactor Consequence Analysis (SOARCA) Project SOARCA Peach Bottom Integrated Analysis Report Volume III				**Doc. No. NUREG-XXXX/SAND2008P-XXXX**	**Doc. Date:** July 2009	**Date Comment Sent:** 20 August 2009
Commenting Individual or Organization: Kevin O'Kula, WSMS				**Phone**/Email: 803.502.9620/kevin.okula@wsms.com	**Resolution by:**	**Phone No**.
No.	**Page**	**Section**	**Type: M – Major Med - Medium Min –Minor E -- Editorial**	**Comment, Question**	**Resolution of Comment**	
1	126 - 137	7.3.1 – 7.3.4	Medium	Figures 63, 65, 67 and 69 show EARLY, CHRONC, and total results for the unmitigated LTSBO sequence, STSBO sequence with RCIC blackstart, unmitigated STSBO sequence, and SST1 source term, respectively. To properly review the offsite consequences of these sequences, tables of the key input parameter values for the EARLY and CHRONC modules are needed. We are interested in site-to-site differences as well as changes in assumptions/inputs from the NUREG-1150 era analysis to the SOARCA analysis.		

Comments/Questions on SOARCA Volume III

COMMENT AND RESOLUTION SHEET						
Document Title: State-of-the-Art Reactor Consequence Analysis (SOARCA) Project SOARCA Peach Bottom Integrated Analysis Report Volume III				**Doc. No. NUREG-XXXX/SAND2008P-XXXX**	**Doc. Date:** July 2009	**Date Comment Sent:** 20 August 2009
Commenting Individual or Organization: Kevin O'Kula, WSMS				**Phone**/Email: 803.502.9620/kevin.okula@wsms.com	**Resolution by:**	**Phone No**.
No.	**Page**	**Section**	**Type: M – Major Med - Medium Min –Minor E -- Editorial**	**Comment, Question**	**Resolution of Comment**	

Comments/Questions on SOARCA Volume IV

COMMENT AND RESOLUTION SHEET						
Document Title: State-of-the-Art Reactor Consequence Analysis (SOARCA) Project SOARCA Surry Integrated Analyses Report Volume IV				**Doc. No. NUREG-XXXX/SAND2008P-XXXX**	**Doc. Date:** July 2009	**Date Comment Sent:** 20 August 2009
Commenting Individual or Organization: Kevin O'Kula, WSMS				**Phone**/Email: 803.502.9620/kevin.okula@wsms.com	**Resolution by:**	**Phone No**.
No.	**Page**	**Section**	**Type: M – Major Med - Medium Min –Minor E -- Editorial**	**Comment, Question**	**Resolution of Comment**	
1	227 - 242	7.3.1 – 7.3.8	Medium	Figures 145, 147, 149, 151, 153, and 154 show EARLY, CHRONC, and total results for the unmitigated STSBO sequence, unmitigated STSBO sequence with TISTGR sequence, mitigated STSBO sequence with TISTGR sequence, LTSBO sequence, unmitigated ISLOCA and SST1 source term, respectively. To properly review the offsite consequences of these sequences, tables of the key input parameter values for the EARLY and CHRONC modules are needed. We are interested in site-to-site differences as well as changes in assumptions/inputs from the NUREG-1150 era analysis to the SOARCA analysis.		

Comments/Questions on SOARCA Volume IV

COMMENT AND RESOLUTION SHEET						
Document Title: State-of-the-Art Reactor Consequence Analysis (SOARCA) Project SOARCA Surry Integrated Analyses Report Volume IV				**Doc. No. NUREG-XXXX/SAND2008P-XXXX**	**Doc. Date:** July 2009	**Date Comment Sent:** 20 August 2009
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No.	**Page**	**Section**	**Type: M – Major Med - Medium Min –Minor E -- Editorial**	**Comment, Question**	**Resolution of Comment**	

October 14, 2009

To: SOARCA Peer Review Team
From: Ken Canavan, EPRI
RE: SOARCA Review Meeting Comments (September 15 and 16, 2009)

The following are comments to date as a result of my individual review of the documents provided as well as participation in the September 15 and 16 meeting. Please note that these are preliminary findings, thoughts and observations for consideration of the panel and authors.

General Comments

1. The SOARCA analysis and report is developed by applying a method to two specific plants Surry and Peach Bottom. The use of 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.

   For example, in the case of Peach Bottom, the drywell does not have a curb inside the drywell and therefore direct containment heating as a result of corium contact with the liner is possible. In other BWR Mark I containments, the liner may prevent or reduce the likelihood of corium contact with the liner.

   As a second example, Surry is chosen for the PWR pilot. In the case of Surry, the location of the interfacing system Loss of Coolant Accident (LOCA) is such that the break will be submerged. This is not the case in many of the other PWRs where specific geometry may result in a non-submerged break. In this case, the plant specific geometry can significant impact the calculated result.

   While an alternative to the current approach or analysis is not recommended or sought by this comment a short discussion of the necessity of the approach as well as the benefits and potentials issues maybe warranted. In addition, sensitivity cases of known issues such as the Surry specific interfacing systems LOCAs may be warranted.

2. In many locations in the report, the facts are provided in the appropriate level of detail. Often these facts represent specifically what was done in the analysis.

What is not always presented is the conclusions that can be drawn from the facts provided or any alternative information that supports the conclusions that are drawn but not stated. The use of affirmative statement and/or any additional evidence that supports the conclusion could be helpful in some instances.

For example, the application of the sequence screening criteria to the Peach Bottom SPAR and plant specific PRA results in no sequences being identified for analysis in SOARCA from the level 1, internal events PRA. However, nothing is concluded as a result of this outcome, which alone is a significant finding, nor is any additional evidence provided that this could or should be expected or that this conclusion may or may not be applied elsewhere. This evidence could include both findings from other BWR PRAs or other anecdotal evidence of why certain sequence groups would not be expected (i.e., plant improvements such as station blackout rule, maintenance rule, improvements in reliability, ATWS rule, hardened vent or others.).

3. An important aspect of this type of analysis is to ensure that it is complete an all aspects and range of variables that can impact the consequences have been considered. During the detailed discussions and question and answer period with the authors it was clear that analysis beyond what was documented in the current 4 volumes had been performed. These discussions and additional analysis, evidence or information should be documented in the reports. So as not to detract from some of the more important points of the analysis, appendices can be used. There are several specific areas which are noteworthy of further consideration, analysis or documentation. These are all in the larger category of completeness and are the treatment of security related events, the treatment of the accident sequence selection and application of the screening criteria and the external event scenarios.

Specific Comments

4. Safety valves and pilot operated relief valves play a significant role in the accident sequences analyzed in SOARCA. Both the successful operation as well as the failure modes under beyond design basis conditions are clearly significant in the analysis. While the failure modes considered in the SOARCA analysis are, in the opinion of this reviewer likely, others with more expertise in the area of safety valves should be consulted.

   One sequence of events included the failure of safety relief valves after 10 cycles of the valve following core damage. At this point in the scenario the safety valves are experiencing approximately 1000 degrees Kelvin fluid flow. This is temperature fluid is significantly beyond the design temperature fluid for the valve and approximately the point where steel will lose its structural strength. It is likely that the postulated failure in the open position will occur. However, the 10 cycles appears relatively arbitrary and is not well supported by the text included in Volume 1. Anecdotal evidence such as the fact that the temperature is

well beyond design and over the point where steel components would lose their structural strength should be noted. In addition, the number of lifts and the number of valves virtually ensures an eventual open valve condition. Lastly, the open valve does not significantly impact the scenarios which is also worthy of emphasis.

5. The impact of the sequence frequency truncations is significant on the outcome of the study. As the study is a consequence study the specific frequency of occurrence of the scenario is not relevant except to choose the most frequent scenario groups to analyze. This is also not well described in Volume 1.

   Volume 1 does describe the scenarios truncation process in overview and also mentions that the consequence analysis does not consider frequency. A simple example is provided that anecdotally support the truncation frequency by stating that sequences an order of magnitude lower would need to be a factor of 10 higher to pose the same risk to the populace. While this is generally true, this can become confusing as the analysis is not supposed to consider scenario frequency or risk only choose those scenarios that are the most frequent. Also, this example points out that if the sequence frequency is indeed an order of magnitude lower and the release much greater (such as two orders of magnitude) the truncation process would not have selected scenarios of the highest risk.

   At this time this reviewer is not suggesting that the truncation process is flawed, only that the text has begged a significant question that remained unanswered. As part of this reviewers tasks will be the attempt to provide any specific scenario groups that maybe missing from the scope of the SOARCA review.

6. As stated in comment 5, the sequence frequency truncation has a significant impact on the results of this consequence study. A sequence truncation frequency of $1x10^{-6}$ per reactor year has been chosen for those sequences groups that contribute to core damage and $1x10^{-7}$ per reactor year for those sequences that contribute to large early release frequency.

   On a generic basis, the BWR accident sequence contributions of a range of initiator and accident sequence groups was estimated and is presented below for consideration. It should be noted that these are general estimates based on the experience of the reviewer. Specific plants will vary within and potentially beyond the range provided below.

   a. BWR LOCAs outside the primary containment. These are a group of accident sequences in two broad categories: Breaks Outside Containment (BOC) and interfacing systems LOCAs. BOC sequences are typified by the failure of Main Steam, Feedwater, HPCI (or HPCS), RCIC, RWCU, and Scram Discharge Valve (generally screened) high pressure lines. ISLOCA sequences are typified by the failure of LPCI Injection line, Core Spray Injection line, Shutdown cooling low pressure lines.

The typical CDF range for both the BOC and ISLOCA is from $1x10^{-9}$ to $5x10^{-7}$ per year. While these sequences would indeed screen from the CDF perspective, the upper range includes some sequences that would not screen from the LERF perspective. This reviewer does not suggest that these sequences do not screen using the criteria rather that the upper end of the range does overlap the criteria.

b. BWR Anticipated Transient Without Scram (ATWS) Events. ATWS is not generally modeled as an initiating event; rather, ATWS is modeled as a subsequent failure of the RPS following any transient event. The failure probability of RPS can be estimated based on the information in NUREG/CR-5500 (Vol. 3). Common Cause failure of the mechanical portions of the RPS is approximately $2.1x10^{-6}$ per demand. Common Cause failure of the electrical portions of the RPS is approximately $3.7x10^{-6}$ per demand. However, failure of the electrical portion of the SCRAM can credit alternate SCRAM methods (e.g., alternate rod insertion).

   The total non-mitigated (CDF) for ATWS events is approximately $1x10^{-7}$ to $3x10^{-7}$ per year. The ATWS sequences typically result in containment failure prior to core damage however there is typically no direct containment bypass (i.e., the releases are to the reactor or auxiliary building). Wetwell failures would result in scrubbed release. Consideration should be given the potential inclusion of these scenarios in the study since while the sequence frequency is typically lower than the CDF truncation the scenarios maybe associated with releases and are generally above $1x10^{-7}$ per year.

c. BWR Other Containment Bypass Events. In general, other containment bypass events include those scenarios where containment is bypassed (i.e. isolation is failed) independent if the initiating event. These include but are not limited to
   i. Failure to isolate MSIV paths
   ii. Failure to isolate Drywell sump lines (not strictly a bypass)
   iii. Failure to isolate Containment vent paths (e.g., DW vent and purge lines) (not strictly a bypass)
   iv. The transient and LOCA initiators all challenge the scram system with subsequent failure to isolation or pre-existing containment bypass.

   These sequences groups typically have a CDF lower that $1x10^{-6}$ per year and a bypass or radionuclide release frequency of less than $1x10^{-7}$ per year. (Typically the pre-existing failure of containment due to isolation of or other large failure is less than $1x10^{-3}$ per demand).

d. BWR Other Potential Sequence Groups With High Release Potential. There are other sequences groups that are not part of the groups discussed above that have the potential for significant releases. These include:
    i. Excessive LOCA with vapor suppression failure which is has an estimated frequency of occurrence of approximately $1x10^{-8}$ per year. This is significantly lower than the SOARCA truncation limits.
    ii. LOCAs with vapor suppression failure which is also estimated at $1x10^{-8}$ per year which is significantly lower than the SOARCA truncation limits.

Appendix C Comments on SOARCA Document Description Submitted to the SOARCA Liaison following March 2010 Meeting

email of March 18, 2010
from Ken Canavan

General Report Comments

1. 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.
2. 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.
3. In addition, Executive Summaries are also not well utilized. Additional care could make them more effective.
4. Seismic research issues and the treatment of seismic have the general impression that their contribution would be a foregone conclusion. The area of seismic sequence development is an area where much research is being performed. It is likely the 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.
5. 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 Operating 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.

Cadareche, March 8th, 2010

Comments by B. Clément on revised SOARCA documentation

It is the opinion of the reviewer that the staff addressed adequately most of the previous comments. Only few new comments are given below. Comments #4 and #5 call for some additional work that, if feasible within the constrained time frame, would improve the report.

No editorial comments are given.

1. 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.

2. Synthesis report § 2.1

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.

3. Synthesis report, SG Induced failures

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.

4. Synthesis report, RPV lower head failure

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.

5. Surry analysis – hydrogen burns

This comment refers to the presentation made by KC Wagner at the last meeting<. 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 σ criterion for flame acceleration and the λ criterion for detonation. Those are given in the following document: "W. BREITUNG & 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"

Below is a compilation of comments from Jeff Gabor
Emails from March 17, 2010

Comments on State-of-the Art Reactor Consequence Analysis Project
MELCOR Best Modeling Practices – Revision 1 - 2/15/2010

1. Overall, a good summary of the MELCOR modeling.
2. I believe Dr. Henry previously identified this, but it would be good to include a discussion of the differences between a BWR and PWR core. This could be added to Section 3.1.3 and simply explain the differences (channel boxes, etc). and provide some discussion of their impact.
3. 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.
4. Section 3.1.1.5 – I would also recommend some discussion of structures in the lower plenum (inst 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.
5. Section 3.1.1.6 – I would recommend some discussion of the impact of structures in the cavity area on debris spreading and cooling.
6. Section 3.1.1.7 – I recommend an explanation of why they assume a PWR valve will fail at a cumulative failure probability of 50% and a BWR valve at 90%.
7. Section 3.1.3.1 – This section needs to discuss Drywell shell failure. Section 4.3 even points here for such a discussion.
8. Section 4.2 – For completeness, DCH in a BWR should be discussed and reasons for it being a low threat included.

Comments on State-of-the Art Reactor Consequence Analysis Project
Appendix A – Peach Bottom Integrated Analysis - Revision 1 - 2/15/2010

1. Overall, good document on the Peach Bottom evaluation. Strong technical basis provided for assumptions and other positions.
2. The end of 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 there. It also points to section 4.7.2, which does not provide any details 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 pathway associated with failure mechanism.
3. Section 5.2 – LT SBO 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.

Additional Comments on SOARCA Report

David Leaver

March 12, 2010.

1. My comment 8a. (October 5, 2009 comment set) 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.
2. 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.
3. 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.
4. Page 68 of the Summary report still says that risks are calculated to 100 miles.
5. 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."
6. 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
7. Summary report, page 22: fourth bullet, frequency range is 1E-7 to 5E-7, not 8E-7.
8. 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.

9. 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.
10. 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 nonevacuating cohort. I could not find such a paragraph in the Executive Summary.
11. 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.
12. 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?

Email of March 15, 2010

Dear Karen:

...

My only other contribution is to suggest you number the executive summaries of the SOARCA reports when they exceed 10 pages in length.

…

Please advise if you desire more from me.

John D. Stevenson
Consulting Engineer
FCSU Corporate Center
6611 Rockside Road, Suite 110
Independence, OH 44131
Phone: 216-447-9440
Fax: 216-446-0514
jstevenson4@earthlink.net

(Sent by email from Jacquelyn Yanch, Mar. 16, 2010)

Questions for the Off-Site Consequences Team:

I am interested in information about the following aspects of the SOARCA study:

(i) dose-rates anywhere, any time, for any of the accident scenarios, to residents in any of the different zones. [I'd like to get an idea of the impact of relocation in terms of dose-rates versus stay-away time - so any additional details from any aspect would be very useful.]

(ii) numbers of people evacuating
[I'd like to look at the fraction of the state population that is evacuated.]

(iii) how long evacuations will last under the different accident scenarios.

(iv) ^{137}Cs vs ^{134}Cs levels (likewise in any zone for any accident, etc. - I want to get an idea of how long the elevated dose-rates will last). Total Cs fractions are given in the document but the two isotopes have different decay times.

Appendix D Memo Providing Guidance on SOARCA Issues

MEMO

Re: Guidance on SOARCA Issues as Requested in the March 2-3, 2010 Meetings

To: SOARCA Team

Through: S. P. Burns

From: K. Vierow
Chair, SOARCA Peer Review Committee*

Date: April 9, 2010

During the March 2-3, 2010 meetings on the SOARCA project, issues arose for which the SOARCA Peer Review Committee members were requested to provide guidance to support post-meeting work efforts. As guidance was requested prior to completion of the Committee's final report, responses are being transmitted in memo format. There was no attempt to arrive at a consensus or influence any individual peer reviewer's opinions.

The Committee members' comments and suggestions are provided below in italic font for consideration by the SOARCA team.

1. Does the Committee have recommendations on how the information regarding dose limits presented by Jacquelyn Yanch may be included in the SOARCA report?

   Jacquelyn Yanch and David Leaver have provided the following comments.

   Use of the current "return home" (i.e., long-term habitability) dose limits certainly seems to be "state-of-the-art" since the individual states and government agencies all use dose limits that are in a similar range. Therefore the SOARCA study is, indeed, reflecting the state of the art.

   However, the fact remains that all of these dose limits (i) are based on very poor data, and (ii) are low in terms of doses and dose rates we currently receive in other applications (e.g. medical doses and elevated natural background areas). For example, the return home dose limit of 500 millirem per year in most states compares with doses from typical computed tomography (CT) scans in excess of 500 millirem, with over 70 million CT scans per year now being performed in the U.S. The average dose rate represented by the return home criterion of 500 millirem in one year is below the natural background dose rate in many parts of the world, and is less than a factor of two above natural background dose rates in the United States.

   There is the concern that society will struggle to try to meet these dose limits by trading off important activities related to returning home, accessing contaminated land, etc. This trade-off might make sense if we were confident we knew the biological effect of these

doses and dose rates, but we are far from confident and in fact the data are associated with very large uncertainties. Thus, one of the consequences of a severe reactor accident might be the chaos (social and economic) that ensues as we try to get life back to normal after the accident. We, as a society, should address this issue before something happens rather than afterwards, especially given the very long latent period of radiation-induced cancer. Since this is potentially a major issue, it would be very good to have some aspect of this highlighted in the SOARCA NUREG.

Jeff Gabor supports inclusion of Jacquelyn Yanch's comments in the SOARCA documents.

Roger Kowieski does not believe that the information regarding the dose limits presented by Jacquelyn Yanch belongs in the SOARCA document.

The U.S. Environmental Protection Agency manual, EPA 400-R-92-001, dated October 1991, provides guidance for implementing the Protective Action Guides (PAGs) by State and local officials during the early phase of a nuclear incident, as well as, long term recovery operations. The PAGs for protection of the public from deposited radioactivity are well documented, and the bases for these values are summarized in this manual. This document recognizes that the relocation is the most effective, and, usually, the most costly and disruptive. It is therefore only applied when the dose is sufficiently high to warrant it. In conclusion, it is suggested that any comments/concerns regarding the recommended PAGs (dose limits) be addressed to the Office of Radiation Programs, U.S. Environmental Agency, Washington DC 20460.

Ken Canavan and Karen Vierow suggest that the information regarding the low magnitude of the dose limits is appropriate in the SOARCA documents but the discussion of dose limit validity belongs elsewhere.

Demonstration that health risks resulting from radioactive releases at the currently accepted dose limits are very low is a compelling argument for the safety of nuclear power plants. However the discussion of the validity of current state-of-the-art dose limits should be directed, as Roger Kowieski recommends, to a party that could address this issue. The SOARCA consequence analyses show that health risks for lower dose limits are also very small in magnitude, negating a need for discussion of an appropriate dose limit within the SOARCA project.

2. Which source of dose conversion factors is most appropriate for use in SOARCA? Is Federal Guidance Report 13 up to date? Is BEIR V best-estimate? Are MACCS2 calculations with other dose conversion factors needed? These calculations could be included in the Uncertainty Study.

   Kevin O'Kula provided the following comments on Federal Guidance Report 13.

   Federal Guidance Report (FGR) 13 dose conversion factors (DCFs) are the most appropriate for use in the SOARCA program. FGR 13 DCFs represent the culmination of considerable work by Keith Eckerman and colleagues at ORNL to maintain a high-

pedigree data set that reflects current models and available bio-dosimetric data. No value is found in applying other, or older, sets of DCF input data in the SOARCA calculations. Therefore, only FGR 13 DCFs are recommended.

Jacquelyn Yanch has provided a comparison of BEIR V and BEIR VII risk estimates in the attached memo.

For the SOARCA study, she recommends the use of BEIR VII risk estimates, rather than those of BEIR V, based on the results of this comparison.

3. Is the comparison of SOARCA calculations using the SST1 source term and the SOARCA source term fair and not misleading?

   Jeff Gabor, Bob Henry, Dave Leaver, Karen Vierow and Jacquelyn Yanch provided the following comments.

   The technologies used in the studies could be compared for the same weather scenarios as this would reflect the accomplishments in radiological source term analysis methodologies over the past 30 years. Chapter 7 in Appendices A and B draws appropriate conclusions from the comparison, specifically, that the SST1 source term is larger than the SOARCA best-estimate source term and that "This reflects improvements in understanding and modeling capabilities developed since the Sandia Siting Study was conducted."

   The health effect risks should not be compared since the Sandia Siting Study consequence analysis methodology and assumptions, unlike SOARCA, are not consistent with today's understanding of radiation health risks.

4. Is SOARCA justified in reporting results at the 50-mile limit?

   Jeff Gabor, Dave Leaver, Kevin O'Kula and Jacquelyn Yanch state that the technical basis for reporting results to a distance of fifty miles is justified.

   Although earlier PRA analyses may have used longer consequence base model distances, the fifty-mile grid is supported by the following arguments:

   - *Current plant license renewal and new design considerations in quantifying cost/benefits of severe accident mitigation alternatives (SAMA) analysis and severe accident mitigation design alternatives (SAMDA) are based on consequences to a 50-mile region. Guidance for performing SAMA analyses is provided in NRC staffed-approved NEI 05-01, Rev. A, Severe Accident Mitigation Alternatives (SAMA) Analysis Guidance Document, and uses a 50-mile basis.*
   - *The Ingestion Planning Zone (IPZ) around current and proposed plants, and used as a basis for evaluation in Environmental Impact Statements (EISs), is fifty miles.*
   - *In Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission, NUREG/BR-0058, Revision 4, (September 2004), it is stated:*

"In the case of nuclear power plants, changes in public health and safety from radiation exposure and offsite property impacts should be examined over a 50-mile distance from the plant site" (p. 29).

- *The individual risk decreases rapidly with distance and is extremely low a short distance from the site boundary (i.e., well over a factor of 1000 below the NRC latent cancer QHO inside 10 miles). However, reporting individual risk results to 50 miles is reasonable for completeness and to show the trend of decreasing risk with increasing distance.*

These peer reviewers recommend that the current discussion in the Summary Document be augmented to better support the application of the 50-mile basis.

Kevin O'Kula added the following clarification.

The SOARCA analysis, and indeed, a PRA, is concerned with a nuclear plant and its operations, and not just the reactor. It should be noted that this is a study of the full plant response to specific postulated accident conditions.

5. Does the Committee have recommendations on future work for SOARCA?

Jeff Gabor, Bob Henry, Dave Leaver, Kevin O'Kula, Karen Vierow and Jacquelyn Yanch provided the following comments.

a. *Full Level 3 PRA*
The SOARCA has evaluated the scenarios which are the major contributors to risk. In this manner, SOARCA is a partial Level 3 PRA and it has provided much data that would be obtained from a full Level 3 PRA, making a full Level 3 PRA less necessary

The results of a full Level 3 PRA would be specific to the nuclear power plant (NPP) for which it was performed; therefore Level 3 PRA results cannot be applied to other NPPs. Conversely, a SOARCA for one plant provides insights for other plants of the same type. If SOARCA-level analyses for other NPP types are conducted and the results do not change greatly, then a full Level 3 PRA can be considered unnecessary for achieving the goals of SOARCA Project.

Ken Canavan goes further to suggest that, as part of future work, the SOARCA team consider a partial or full Level 3 PRA.

There is the possibility that certain accident sequences, while not-dominant from a frequency basis, may have relative high risk due to high consequence. While these sequences may not dominant the risk, in terms of frequency and consequence, they could be contributors. Collections of several lower order sequences, while individually non-dominate, could have higher consequence than SOARCA evaluated and could contribute to the risk collectively. While SOARCA did indeed capture the most likely sequences and accurately capture the consequence from these sequences, the primary issue with consequence analyses of this type is that it is difficult to

demonstrate completeness. A Level 3 PRA performed for the SOARCA plant could have the benefit of reduced resources (due to work performed for SOARCA) as well as the benefits of validation of the SOARCA approach and demonstration of completeness. For these reasons, a Level 3 PRA for the SORCA plant might have some value.

Bernard Clement is of the opinion that full Level 3 PRAs are of great interest, independently of SOARCA.

b. *SOARCAs for other plants*
These peer reviewers recommend that SOARCAs be conducted for other NPP types with different containment designs. The change in results from one NPP type to another should be investigated. As mentioned above, if the results do not vary greatly, a full Level 3 PRA would probably be unnecessary.

Regarding the selection of plant types, the remaining plants from the five considered for NUREG-1150 reactors or a down-selection from the eight reactor types that the NRC originally considered would be reasonable.

John Stevenson recommends an evaluation of plant foundation conditions.

Plant foundation conditions at the Surry Site indicate the potential for liquification and consolidation due to earthquake at the SOARCA very low earthquake probabilities of exceedence. This may be considered as a-follow-on SOARCA effort.

c. *Statement on the scope of SOARCA*
Several consequences of a severe accident have not been evaluated within the context of the SOARCA project. These include land contamination, economic losses and recovery costs. A statement should be made in the SOARCA documentation that they are beyond the scope of SOARCA.

Other than as commented in items 1 and 5, Ken Canavan concurs with the memo.

Other than as commented in item 1, Roger Kowieski concurs with the memo.

Bruce Mrowca has not provided an opinion.

John Stevenson wrote the following statement, which is applicable to this memo except for item 5.b. "For the other areas where you have requested input from the Peer Group, I consider them outside my areas of expertise so I am not commenting on them."

*SOARCA Peer Review Committee Members:
Ken Canavan
Bernard Clement
Jeff Gabor

Robert Henry
Roger Kowieski
David Leaver
Bruce Mrowca
Kevin O'Kula
John Stevenson
Karen Vierow
Jacquelyn Yanch

Memorandum:

Re: Choice of Risk Estimates (Is BEIR V "best estimate"?)

To: Karen Vierow, Chair, SOARCA Peer Review Committee
Shawn Burns, SOARCA Study Team

From: Jacquelyn C. Yanch
Member, SOARCA Peer Review Committee

Date: 9 March 2010

The current analysis of late cancer fatality risk in the SOARCA study is based on use of BEIR V (1990) risk estimates. BEIR V estimates of radiation-induced cancer risk do not incorporate a low dose, low dose-rate effectiveness factor (DDREF) which would make the risk estimates applicable to situations where individuals are exposed continually and at a low dose rate. On the other hand, risk estimates published in BEIR VII (2006) do incorporate a DDREF and use a value of 1.5. That is, the risk estimates generated from a review of the Life Span Study population (A-bomb survivors) are reduced (divided by 1.5) to account for the sparing effect that might be expected if the same doses were delivered at a lower dose-rate.

The BEIR VII document asserts that the risk estimates from BEIR V and BEIR VII are similar, but only if the DDREF value of 1.5 is applied to the BEIR V data. Comparisons of BEIR V and BEIR VII estimates are made in Tables 12-8 and 12-9 of the BEIR VII report (pages 282-3); these tables are reproduced below. [To facilitate comparison with BEIR VII, the BEIR V estimates are shown as published, and then again divided by a DDREF of 1.5; these modified data appear in parentheses.]

Also shown in Tables 12-8 and 12-9 are the risk estimates generated by the ICRP, the EPA, and by UNSCEAR. The ICRP and EPA estimates include a DDREF of 2. UNSCEAR and BEIR V include no DDREF and neither document provides guidance for modifying the risk estimates to apply to situations involving low doses and/or low dose-rates.

Recommendation:
Given that exposure to radiation following a reactor accident will generate low doses delivered at low dose-rates, the use of a DDREF is warranted. For the SOARCA study, therefore, use of BEIR VII risk estimates, rather than those of BEIR V, is recommended. This recommendation is based on (i) the incorporation of a DDREF in BEIR VII, making the risk estimates more applicable to the post-accident irradiation scenario, and (ii) the 'best estimate' nature of the BEIR VII estimates which are based on an additional twelve years of follow-up of the Life Span Study population (relative to BEIR V). Use of a DDREF is also consistent with the approach adopted in Federal Guidance Report 13 in which a DDREF of 2 is used in the generation of risk estimates.

International Commission on Radiological Protection (1999) Risk Estimation for multifactorial diseases. Ann. ICRP 29:1-144.
United Nations Scientific Committee on Effects of Atomic Radiation (2000) Sources and Effects of Ionizing Radiation. UNSCEAR Report to the General Assembly.
Environmental Protection Agency (1994) Estimating Radiogenic Cancer Risks, EPA Report 402-R-93-076. Washington DC: Environmental Protection Agency.
National Research Council (1990). Health Effects of Exposure to Low Levels of Ionizing Radiation (BEIR V). Washington DC: National Academy Press.
National Research Council (2006). Health Effects of Exposure to Low Levels of Ionizing Radiation (BEIR VII). Washington DC: National Academy Press.
EPA (1999) Cancer Risk Coefficients for Environmental Exposure to Radionuclides. Federal Guidance Report No.13.

Tables from BEIR VII report (National Academy of Sciences) 2006, pages 282 and 283.

TABLE 12-8 Comparison of BEIR VII Lifetime Cancer Mortality Estimates with Those from Other Reports

Cancer Category	BEIR V[a] (NRC 1990)	ICRP[b] (1991)	EPA[b] (1999)	UNSCEAR[c] (2000)		BEIR VII[d]
Leukemia[e]	95 50	56	50			61
All cancer except leukemia (sum)	700 (460)	450	520			
All solid cancers (sum)				1150, 780 1400[f], 1100[f] (520)		510
Digestive cancers	230 (150)					
Esophagus		30	12	30, 60	(25)	
Stomach		110	41	15, 120	(18)	22
Colon		85	100	160, 50	(75)	61
Liver		15	15	20, 85	(20)	16
Respiratory cancer	170 (110)					
Lung		85	99	340, 210	(160)	210
Female breast[g]	35 (23)	20	51	280, 65	(43)	37
Bone		5	1	—		
Skin		2	1	—		
Prostate[g]						5
Uterus[g]						3
Ovary[g]		10	15			12
Bladder		30	24	40, 20	(22)	25
Kidney		—	5	—		
Thyroid		8	3	—		
Other cancers or other solid cancers[h]	260 (170)	50	150	280, 180	(160)	130

NOTE: Excess deaths for population of 100,000 of all ages and both sexes exposed to 0.1 Gy.

[a]Average of estimates for males and females. The measure used was the excess lifetime risk; unlike other estimates in this table, radiation-induced deaths in persons who would have died from the same cause at a later time in the absence of radiation exposure are excluded. The estimates are not reduced by a DDREF, but parentheses show the result that would be obtained if the DDREF of 1.5, used by the BEIR VII committee, had been employed.

[b]Except for the EPA breast and thyroid cancer estimates, the solid cancer estimates are linear estimates reduced by a DDREF of 2.

[c]Average of estimates for males and females. Except where noted otherwise, estimates are based on the attained-age model. The first estimate is based on relative risk transport; the second on absolute risk transport. The estimate in parentheses is a combined estimate (using the same weights as used by the BEIR VII committee applied on a logarithmic scale) reduced by a DDREF of 1.5, although these were not recommendations of the UNSCEAR committee.

[d]Average of the committee's preferred estimates for males and females from Table 12-5B.

[e]Estimates based on a linear-quadratic model.

[f]Estimates based on age-at-exposure model.

[g]These estimates are half those for females only.

[h]These estimates are for the remaining solid cancers.

TABLE 12-9 Comparison of BEIR VII Lifetime Sex-Specific Cancer Incidence and Mortality Estimates with Those from Other Reports

	Males			Females		
Cancer Category	BEIR V[a]	UNSCEAR[b]	BEIR VII[c]	BEIR V[a]	UNSCEAR[b]	BEIR VII[c]
Incidence						
Leukemia[d]	NA	50	100	NA	50	72
All solid cancer	NA	1330, 1160 (740) 2600,[e] 1700[e]	800	NA	3230, 1700 (910) 3800,[e] 2100[e]	1310
Mortality						
Leukemia[d]	110	50	69	80	60	52
All cancer except leukemia (sum)	660 (440)			730 (490)		
All solid cancers (sum of sites)		710, 620 (380) 900,[e] 900[e]	410		1580, 930 (660) 1900,[e] 1300[e]	610

NOTE: Excess deaths for population of 100,000 of all ages exposed to 0.1 Gy.

[a]The measure used was the ELR; unlike other estimates in this table, radiation-induced deaths in persons who would have died from the same cause at a later time in the absence of radiation exposure are excluded. The estimates are not reduced by a DDREF, but parentheses show the result that would be obtained if the DDREF of 1.5, used by the BEIR VII committee, had been employed.

[b]Except where noted otherwise, estimates are based on the attained-age model. The first estimate is based on relative risk transport; the second on absolute risk transport. The estimate in parentheses is a combined estimate (using the same weights as used by the BEIR VII committee applied on a logarithmic scale) reduced by a DDREF of 1.5, although these were not recommendations of the UNSCEAR committee.

[c]Estimates are from Tables 12-6 and 12-7, and are shown with 95% subjective confidence intervals.

[d]Estimates based on a linear-quadratic model.

[e]Estimates based on age-at-exposure model.

Appendix E Memo on Uncertainty Quantification and Sensitivity Analysis

MEMO

Re: Guidance on the SOARCA Uncertainty Quantification and Sensitivity Analysis

To: SOARCA Team

Through: S. P. Burns

From: K. Vierow
Chair, SOARCA Peer Review Committee*

Date: April 9, 2010

The SOARCA Team presented plans for an Uncertainty Quantification and Sensitivity Analysis to the SOARCA Peer Review Committee on March 3, 2010. Comments from Committee members and suggestions regarding this effort are provided below for consideration by the SOARCA team. There was no attempt to arrive at a consensus or influence any individual peer reviewer's opinions.

Bob Henry, David Leaver, Kevin O'Kula and Karen Vierow have provided input to this memo and concur with the sections that they did not compose.

1. Of the two methods presented for quantifying uncertainty, the "Inner" Weather Loop method is the appropriate method for evaluating the SOARCA results and for comparing with the previous NRC studies. A few sequence results should be explored through the "Outer" Weather Loop method to illustrate the influence of uncertainty in weather conditions at the time of the release.

   The inner loop method preserves the perspective that the SOARCA source term is smaller and later in release to the environment than source terms used in previous risk work. In this manner, the modeling advancements and new insights from experimental testing of the past twenty years are reflected. The outer loop method provides results that are more influenced by the effects of site-specific weather. While the impact of site weather is important, it will statistically change little from year to year, and is not changeable through any SOARCA-based understanding or insights. Therefore, the inner loop method should better suit the objectives of the SOARCA project in discerning improved understanding of the risk from Nuclear Power Plant operation.

   The outer loop, however, provides a mechanism for looking at more limiting weather conditions. By performing a limited number of sensitivity analyses with the outer loop method in addition to analyses by the inner loop method, the SOARCA project can provide some insights when considering the uncertainty of both the source term and the weather.

2. The Uncertainty Quantification and Sensitivity Analysis study is essential to the credibility of the SOARCA project and should be documented as part of the SOARCA NUREG report, or as a stand-alone supporting reference.

3. The Uncertainty Quantification study is in its early stages of planning and was not available for Committee review. Nonetheless, the uncertainty analysis is an integral part of the SOARCA project, and the analysis could be regarded as incomplete if there is not an attempt to address uncertainty. The members of the Peer Review Committee concurring with this memo request the opportunity to review the uncertainty quantification effort. Parameter selection and parameter distributions require particular care. Updates as well as the final set to be used in the Uncertainty Quantification study are requested.

Ken Canavan, Bernard Clement, Jeff Gabor and Jacquelyn Yanch concur with the memo as it is written above.

Roger Kowieski stated that he concurs with the memo as written above and that "the Uncertainty Qualification study is essential to the credibility of the SOARCA project."

Bruce Mrowca has not provided an opinion.

John Stevenson wrote the following statement, which is applicable to this memo. "For the other areas where you have requested input from the Peer Group, I consider them outside my areas of expertise so I am not commenting on them."

*SOARCA Peer Review Committee Members:
Ken Canavan
Bernard Clement
Jeff Gabor
Robert Henry
Roger Kowieski
David Leaver
Bruce Mrowca
Kevin O'Kula
John Stevenson
Karen Vierow
Jacquelyn Yanch

MEMO

Re: Guidance on the SOARCA Uncertainty Analysis Plan

To: SOARCA Team

Through: S. P. Burns

From: K. Vierow
Chair, SOARCA Peer Review Committee*

Date: December 22, 2010

As an action item from the Oct. 26-27, 2010 meeting between the SOARCA team and the SOARCA Peer Review Committee, comments and guidance on the Uncertainty Plan presented in the following reference are attached.

Mattie, P. D., D. A. Kalinich, S. P. Burns, State-of-the-Art Reactor Consequence Analysis (SOARCA) Project: Uncertainty Analysis Plan Recommended by Sandia Technical Staff – DRAFT, prepared by Sandia National Laboratories, Oct. 19, 2010.

*SOARCA Peer Review Committee Members:
Ken Canavan
Bernard Clement
Jeff Gabor
Robert Henry
Roger Kowieski
David Leaver
Bruce Mrowca
Kevin O'Kula
John Stevenson
Karen Vierow
Jacquelyn Yanch

Some Comments on SOARCA Uncertainty Analysis Plan

B. Clément

December 22, 2010

General

As explained, the effort is limited due to time and resource constraints. This is understandable but will induce limitations in the conclusions that will be drawn from the exercise. It can be expected that the applicability of the methodology is, if not fully demonstrated, at least illustrated on the specific example that is treated. Concerning the specific sequence that will be treated it is expected that the exercise will show the effect of the most influential parameters. It will however not be possible to extrapolate in general the conclusions to other accidental sequences and other reactors.

The exercise is mainly based on a statistical treatment of the propagation of uncertainties of models parameters. This is probably the best way to proceed when using complex simulation tools such as MELCOR and MACCS. It should however be recognised that some epistemic uncertainties are not taken into account with this method, especially when some physical phenomena of potential importance are not modelled. When this is known, it should be clearly stated. An example is given in page 43 for CH_3I.

In the following, we concentrate on some aspects of MELCOR uncertain parameters and distribution.

MELCOR Uncertain Parameters and Distribution

The Zr melt breakout temperature is certainly an important uncertain parameter. Its influence will be evaluated for the chosen scenario but it should be kept in mind that the relative influence of this parameter should largely differ for other scenarios.

Molten clad drainage rate: it is worthwhile to study this parameter although it is probably less important than the previous one.

Fuel failure criterion is also an important parameter. There is certainly some correlation between its influence and the influence of Zr melt breakout temperature. The importance of fuel-cladding interactions will depend on the latter and in turn affect the temperature and time at which there will be a transition from rod-like geometry to debris. A kind of correlation between the probability density functions of these parameters could in principle be established by looking at the results of MELCOR analyses of some experiments such as for instance Phebus FP. If this has not already been done, it should however be recognised that it would require quite a lot of work probably not manageable by the Project.

The wording about hydrogen ignition criteria is unclear as it is said both “no consideration currently given to possibility of the absence of an ignition source” and “accumulation of H2 due to the absence of ignition source is credible”.

Concerning chemical forms of iodine and cesium, the relative probability of the five different combinations is not justified. Concerning the fraction of gaseous iodine (that can be considered as I_2 as proposed) the value recommended by NUREG-1465, that is supposed to be best-estimate is 0.05. Unless there are good arguments, there should be the central value and not the upper bound. Concerning cesium, the partition between CsOH, CsI and Cs_2MoO_4 might have an impact on releases of both cesium and iodine. It is recommended to wait for the first results of the study to decide if other partitions (increased number of combinations) would be useful.

Concerning the dynamic and agglomeration shape factors, not only the hygroscopic effects will cause the particles to tend towards being spherical but also the agglomeration processes (see e.g. M.P. Kissane: On the nature of aerosols produced during a severe accident of a wter-cooled nuclear reactor, Nuclear Engineering and Design 238 (2008) 2792-2800).

SOARCA Uncertainty Analysis Plan: Peer Review Comments by Jeff Gabor

Based on *Uncertainty Analysis Plan Recommended by Sandia Technical Staff* – DRAFT, 10/19/2010

1. Technical justification needs to be documented for the selection of each uncertainty parameter. Included should be a discussion of the best estimate value along with the basis for the range of uncertainty and the appropriate distribution. Technical references should be included for each parameter.
2. Many of the MELCOR uncertain parameters depend on one and other. Correlations between various parameters should be developed to represent these dependencies.
3. In-core failures of instrument tubes have been raised by Dr. Henry and need to be addressed in the SOARCA documentation. If uncertainties exist as to the potential for enhanced radionuclide release, these should be included in the uncertainty analysis.
4. The potential for Main Steam Line creep failure is not being considered as an uncertain parameter. The MSL rupture area for such a failure would likely be large and probably does not need to be evaluated as an uncertain parameter. I recommend that this uncertain parameter be replaced with a parameter that more directly relates to MSL creep failure.
5. Failure of both sets of railroad doors is included in the uncertainty, however, failure of the blowout panels at the top of the refueling floor are not included. Given the complexity of how the reactor building will respond to a containment failure, it is recommend that the RR door failure parameter be replaced with a more simplified parameter representing the potential decontamination factor for the entire reactor building.
6. The success and timing of operator actions should also be considered in the uncertainty analysis. These actions include opening of the SRVs earlier in the BWR event along with initial level control using RCIC.

Karen Vierow

From: Henry, Robert E [Henry@Fauske.com]
Sent: Monday, November 22, 2010 2:39 PM
To: Karen Vierow
Cc: Henry, Robert E
Subject: My Comments on the SOARCA Uncertainty Study
Attachments: image001.jpg

Karen:

My comments on the uncertainty study closely parallel my comments related to the information that we have yet to receive to quality the code. The uncertainty study only has substance when the variations in the mechanistic models are based on the uncertainty variations that have been derived by comparing the mechanistic models with various experiments. Without this, there is there is no technical basis for the variations that are used in the evaluations. In the framework of the uncertainty study that was sent to us, there were a number of parameters that were used to represent variations in the accident progression, but to date, no technical basis has been provided for any of these. Out of frustration, I Googled and found a presentation entitled "MELCOR Code Development Status, Code Assessment and QA". Under the heading of MELCOR 2.1 Assessment Matrix they discuss the TMI-2 accident as well as the following experiments:

- LOFT-FP2,
- PHEBUS,
- IIST,
- BACCHUS,
- LHF,
- OLHF,
- VERCORS,
- CVTR,
- HDR,
- NUPEC,
- Marviken,
- CSTF,
- PANDA,
- ABCOVE,
- SUPRA, and
- DEMONA.

We have requested information relating to how the model parameters are varied as a function of these, or other benchmarks, if at all. More to the point, we have asked for these so that we can understand how the uncertainty variations are being quantified. The MELCOR best practices document that was given to the committee identifies MELCOR 1.8.6 as the code used to assess radiological releases to the environment. The assessment matrix that lists the experiments given above identifies MELCOR 2.1. Is this the difference that we are confronting? Have these benchmarks not been performed with 1.8.6 but are available for version 2.1? The committee somehow must be able to develop an appreciation of the extent of benchmarks that exist for the code that the NRC is using in this evaluation. Without the identification of the technical basis, neither the "best estimate" calculations nor the "uncertainty analysis" have any basis.

As always, let me know if you have any questions.

Regards,
Bob

Robert E. Henry, PhD
Senior Vice President

FAUSKE & ASSOCIATES, LLC
WORLD LEADER IN NUCLEAR AND CHEMICAL PROCESS SAFETY

Tel: +1 630 887 5201 *16w070 83rd Street*
Mob: +1 630 251 8648 *Burr Ridge, IL 60527 USA*
Fax: +1 630 986 5481 *Main: +1 877 FAUSKE1*
www.fauske.com

Karen Vierow

From: NTHMC@aol.com
Sent: Monday, December 06, 2010 9:37 AM
To: Karen Vierow
Subject: Re: FW: list of unresolved issues from SOARCA peer reviewers

Karen,

I have no comments on the Uncertainty Analyses.

Roger Kowieski

Review of SOARCA Uncertainty Analysis Methodology

David Leaver

December 20, 2010

This note is to document my review of the SOARCA preliminary methodology for uncertainty analysis. The preliminary methodology was primarily defined in two draft reports: (1) "Uncertainty Analysis Plan Recommended by Sandia Technical Staff – DRAFT", and (2) "Evaluation of Distributions Representing Important Non-Site-Specific Parameters in Off-Site Consequences." The peer reviewers also had the benefit of a 1 1/2-day meeting with SNL and NRC staff to discuss the methodology.

Specific comments on the two reports are provided below. Several general comments are as follows:

- Section 4 of the first report notes that for some uncertain parameters there is a limited or in some cases no technical basis documented. This must be addressed in the final uncertainty report, even if the technical basis is qualitative and subjective. Some examples are discussed below. The basis should address the upper and lower bounds and the nature of the distribution.
- There are significant editorial errors in the first report.
- The uncertainty analysis methodology is attempting to address uncertainty in an integrated manner within the context of the core damage progression model and the consequence analysis. This is a worthwhile approach and it is innovative, but has the problem that dependencies among uncertain parameters may be hard to discern in some cases. The approach differs from traditional PRA-based approaches to uncertainty analysis which apply the Level 1 and Level 2 logic models, split fractions, and the like to identify these dependencies. In light of this, it is suggested that the SOARCA project consider use of traditional sensitivity studies as a supplement and sanity check for the integrated uncertainty analysis and as a way to provide insight on potential dependencies.

Specific comments are as follows:

<u>Uncertainty Analysis Plan Recommended by Sandia Technical Staff – DRAFT</u>

- Page 41 of Appendix A, "Peach Bottom Integrated Analysis," indicates that station blackout leads to loss of room cooling for the RCIC and HPCI corner rooms, and that the operators are directed to block open doors to these rooms to facilitate cross ventilation so as to slow the rate of room heatup. These actions were assumed to successfully prevent system isolation from high temperature for the maximum period of 4 hours of system operation. Thus there are two uncertain aspects to loss of room cooling: (1) the likelihood of successful operator action to block open the room doors; and (2) the likelihood that these actions prevent system isolation due to high temperature (note that apparently no modeling has been performed on the room heatup). Based on this, loss of room cooling should be considered as an uncertain parameter for the LTSBO uncertainty analysis.

- For the unmitigated LTSBO sequence, operator actions credited are opening one SRV (1 hour) and taking manual control of RCIC (2 hours) and as described on page 43 of Appendix A maintaining water level at 5 in to 35 in (i.e., 16 ft above TAF). These actions are uncertain (or rather, the time at which these actions are taken is uncertain) and thus should be included on the list of uncertain parameters.
- In the paragraph on Main steam line creep ruture area, should the uncertainty in main steam line temperature be considered?
- Fuel failure criterion is an example of a parameter where there is lack of a basis for the characterization of the uncertainty (see general comment 1). The uncertainty proposed is not unreasonable, but some kind of basis is required, even if it is simply a qualitative discussion of how time at temperature can affect the various phenomena that impact fuel mechanical response as the fuel heats up.
- Radial debris relocation time constants is another example of a parameter where there is no basis for the characterization of uncertainty. Also, there are two incomplete sentences at the bottom of this paragraph.
- Figure 4.1-7 has the wrong title.
- The abscissa of Figure 4.1-15 should be labeled (H2 mole fraction?).
- Why is Zr oxidation fraction and amount of H2 generation not an uncertain parameter?
- There should be a basis for the fraction of I2. There is a lot of R&D going on regarding iodine chemistry that could be consulted.
- Indication of the MACCS2 values would be useful for comparison with Tables 4.2-3 and 4.2-4.
- What uncertainty is Figure 4.2-10 reflecting?
- The SOARCA report has 6 cohorts whereas Table 4.2-8 has 5. Perhaps Corhort 6 (non-evacuating public) was left off, but is there not some likelihood that they will actually evacuate with some delay?
- Table 4.2-10: dose threshold for what?

<u>Evaluation of Distributions Representing Important Non-Site-Specific Parameters in Off-Site Consequences</u>

- The spread of dispersion estimates in Figures 2.47 through 2.54 are generally 1.5 to 2 orders of magnitude, and X/Q generally goes as $1/\sigma_y\sigma_z$. This spread is based on the 1995 – 1998 expert data cited in the report (references 1 to 6). This spread seems too high based on what was learned from the more recent comparisons of MACCS predicted X/Q with the LLNL detailed model (maybe a factor of 2 – 3 different). Why not adjust the older, subjective expert data to reflect the more rigorous and recent LLNL comparison?
- To provide a comparison and sanity check on the correlations between dry deposition and wet deposition data, the deterministic particle velocity vs. particle size from the MACCS model should be displayed.
- There should be some discussion and possibly adjustment regarding the fact that ICRP indicates ~5E-4 fatality risk vs. ~1E-3 for the 0.5 quantile in Table 5.1.
- On page 89, what is the exposure time associated with the 100 Gy/hr?

- Figure 6.1 is puzzling. Expert A has 0.0 – 0.05 quantile threshold dose of about 80 rem and Expert I has ~300 rem (seems more like it it me). How can these two "experts" not be at least somewhat close on the 0.0 – 0.05 quantile for threshold dose if the problem was well-posed and they are addressing the same problem? And how is it possible for ~0.8 Sv (80 rem) to cause an early fatality? Would this be for a non-healthy person?

DATE: December 10, 2010

TO: Professor Karen Vierow
Associate Professor
Department of Nuclear Engineering
Texas A&M University
129 Zachry Engineering Center, 3133 TAMU
College Station, TX 77843-3133

FROM: Kevin O'Kula

Uncertainty Analysis Comments

In general, the planned uncertainty analysis is well thought out and should achieve the objectives discussed with the SOARCA Peer Review Group during March and October meetings. With the work that has been presented to the Peer Review Group since 2009, and progress in research and analytical methods since the Sandia Siting Study, the improvement in uncertainty in severe accident consequences should be as important as the reduction in the quantitative mean values obtained for specific risk metrics. The clarity of this message is important and must be made apparent in the final uncertainty analysis report.

General

Most comments in this memorandum will focus on the consequence analysis portion of the uncertainty study. The set of consequence (MACCS2 analysis) uncertain parameters is appropriate considering their respective importance to the overall outcome. Of the set of eleven parameters to be included, i.e., dispersion parameters (σ_y and σ_z), washout model linear coefficient, *deposition velocity*, shielding parameters, early health effects, normal and hotspot relocation, evacuation delay and speed, *groundshine*, *habitability*, the three italicized parameters are expected to have large impacts (at least to this reviewer).

Specific topics follow:

1. **Selection of Peach Bottom Unmitigated Long-Term Station Blackout (PB LTSBO)** – This scenario is a reasonable choice to exercise appropriate parts of the source term – consequence methodology.
2. **Bypassing a Phenomena Importance and Ranking Table (PIRT) Process -** Presentations and discussions made in March, and again in the October meetings noted that a PIRT process was not in the scope of work and subsequently, "**a limited set of key parameters and their distributions was compiled that relies heavily on the best available data and expert judgment."** Given that the SOARCA uncertainty analysis will extend into the first half of CY2011, it is recommended that this position be revisited. In the long run, a targeted PIRT process would be a useful exercise to preclude inadvertent omission of parameters from the uncertainty analysis that later are deemed to be important. The PIRT process could be done in a short-term, expedited manner for both the source term phase and the consequence analysis phase, and documented as appendices to the final report.

K.R. O'Kula, URS Safety Management Solutions LLC/ Phone: 803.502.9620; Email: kevin.okula@wsms.com

3. **Model Uncertainty** – The uncertainty analysis as described by the SNL-NRC SOARCA project team is a parameter uncertainty analysis. It would be very informative to explore the uncertainty of several model options, or whether refinements in a base model helped reduce uncertainty in the overall analysis. NUREG-1855, Volume 1, and Regulatory Guide 1.174 are useful approaches to guide these types of analysis. The two categories that are of particular interest are:
   - **Refinement of Polar Grid in ATMOS** –The previous polar coordinate grid model in MACCS2 (Version 1.13.1) allowed 16 sectors. The new model in WinMACCS2 (V3.5.0 beta) uses 64, based on a suggestion made during the first round of SOARCA meetings in Albuquerque several years ago. It would be a valuable insight to understand if this helped reduce (1) the mean result; and (2) reduced uncertainty about the 50th percentile.
   - **Latent health effect models** – Four approaches have been applied in the SOARCA analysis for calculation of latent cancer fatality risk, LNT, ICRP 104 (truncation of 10 mrem/y), U.S. Background Average (truncation of 620 mrem/y), and the HPS (explanations provided in the SNL-NRC reports). It would be a significant milestone if a limited model uncertainty could be explored for LCF risk for both plants.
4. **Clarity of Discussion in Aleatory and Epistemic Uncertainty Results** – This discussion is very good and needs to be repeated in summary at the end of the uncertainty analysis results. In addition, it is imperative that the study note an improvement in uncertainty with respect to results over earlier work, such as the Sandia Siting Study, and that is presented clearly without being hidden. If a comparison of overall uncertainty is not feasible, perhaps several important uncertain parameters can be compared in to what they were previously, then it would be key that these same parameters be compared.
5. **Definition of Uncertainty Analysis and Sensitivity Analysis** – Section 3.2 of the Uncertainty Analysis plan contains two definitions, i.e., that of uncertainty analysis and sensitivity analysis, that ought to be rewritten in simpler English for the final report. Although this content was in the draft uncertainty analysis plan, there is still too much importance of this section to leave as it stands. The sentence reads:

   > "Closely associated with the characterization of epistemic uncertainty provided by the probability space corresponding to EN3 and the answering of Question Q4 are the concepts of uncertainty analysis and sensitivity analysis, where uncertainty analysis designates the determination of the epistemic uncertainty in analysis results that derives from epistemic uncertainty in analysis inputs and sensitivity analysis designates the determination of the contribution of the epistemic uncertainty in individual analysis inputs to the epistemic uncertainty in analysis results."

6. **Section 2.4 Description of the Probabilistic Analysis Methodology for SOARCA -** This information, while useful, should be published as an appendix in the final report.
7. **Percentile Bounds on Work** – The briefing in October from Patrick Mattie noted that 25th/75th percentile bounds would be used on the CCDFs using the bootstrap method. There may have been a simple explanation for this set of bracketing bounds given during the discussion, such as a requirement on the number of simulations, or other limiting factor, but I don't recall it and the more common set of bounds are 5th and 95th.
8. **Table and Figure Discussion** – The importance of weather trials in the MACCS2 sampling methodology is known but inclusion of this term as an uncertainly parameter in

Table 2.2-1 is not explained although several comments are made throughout Section 2.2. Please clarify in the final report. Examples of scatterplots, SRCs/PCCs as a function of time, and illustration of failure of a sensitivity analysis are illustrated in Figures 3.2.1, 3.2.2, and 3.2.2, respectively, are provided with little if any explanation. While these are meant to be illustrative in the plan, if they are used in the final report, please include sufficient discussion as to what is being shown in the figure of interest.

9. **Version of codes being applied** – Mattie's presentation indicated the versions of MELCOR, MELGEN, and MACCS2/WinMACCS being applied, but this is not documented consistently in the reports received to date. It is recommended that the final uncertainty analysis reports state directly the codes and versions being applied to support the final uncertainty study.
10. **General Comment on "Evaluation of Distributions Representing Important Non-Site Specific Parameters in Off-Site Consequence Analyses" and Implementation in SOARCA** – The manner in which deposition velocities are input for the nine physicochemical groups used in the MACCS2 Peach Bottom or Surry plant analysis is still unclear. Specifically, the manner in which each of the nine radionuclides groups is associated with a RDPSDIST00X bin and how the median diameter bins would be distributed for a given radionuclide group (example shown in Figure 1). While a brief discussion on this topic was given in the March briefing, this topic remains an issue because this SOARCA study apparently uses information from an updated expert elicitation, "Evaluation of Distributions Representing Important Non-Site-Specific Parameters in Off-Site Consequence Analysis". Granted larger particle size deposition velocity will be site-independent and behavior should follow Stokes Law, the deposition velocity for smaller aerosol particle size groups (say for $d_p \lesssim 1$ μm) is site-dependent. This is because smaller particles will interact with regional ground cover such as the tree canopy and other vegetation. Consequently, field testing over forested sites typically shows higher deposition velocities compared to those with relatively little surface cover. Admittedly, this issue is perhaps a minor one for the SOARCA study as both sites are in the Eastern U.S. Nonetheless, a discussion on the specifics for implementation of final values of the deposition velocity by radionuclide bin should be provided in the final study as well as in the uncertainty analysis.

```
Example Usage:
*
* PARTICLE SIZE DISTRIBUTION OF EACH ELEMENT GROUP
*
*            0.001   0.01   0.02  DEPOSITION VELOCITY OF EACH GROUP (METERS/SECOND)
*
RDPSDIST001  0.1     0.8    0.1
RDPSDIST002  0.1     0.8    0.1
RDPSDIST003  0.1     0.8    0.1
RDPSDIST004  0.1     0.8    0.1
RDPSDIST005  0.1     0.8    0.1
RDPSDIST006  0.1     0.8    0.1
RDPSDIST007  0.1     0.8    0.1
RDPSDIST008  0.1     0.8    0.1
RDPSDIST009  0.1     0.8    0.1
```

Figure 1. Allocation fraction input illustration from page 5-27 of MACCS2 User's Guide

(comment by K. Vierow: this memo is authored by Dr. John Stevenson)

October 29, 2010

Dear Karen:

I would like to comment on the text of Section 2.1 of the "Uncertainty Analysis Plan Recommended by SANDIA Technical Staff," Draft dated 10/19/2010 which provides justification of the use of a seismic induced Station Beach Out, SBO as the scenario to be evaluated in the SOARCA as it relates to other NPP sites in the U.S.

With respect to seismic being the greatest natural hazard, there are a few NPP sites in Florida and along the Gulf of Mexico coast where the potential for a LOCA associated with a Category 5 hurricane and associated storm surge flooding may exceed the risk from an earthquake at the 10^{-6} to 10^{-7}/yr probability of exceedence level. Less than 10 percent of all NPP sites in the U.S. may be in this category leaving more than 90 percent of NPP sites being dominated by an earthquake.

With respect to those NPP sites dominated by earthquakes, perhaps 70 percent seismic induced SBO would dominate the risk of an unmitigated LOCA . The other 30 percent might be expected to be dominated by early containment failure as a result of seismic induced building foundation failure (i.e. liquification or consolidation) of the containment or adjacent structure foundation due to an earthquake at the 10^{-6} or 10^{-7}/yr probability of exceedence level.

I am also providing comments relative to the one and half day meeting held at NRC offices 26 and 27 October 2010. Again, the selected scenario for the limiting event discussed at the meeting was an extremely low probability of exceedence seismic event (i.e. less than 10^{-6} or 10^{-7}/yr probability of exceedence earthquake) causing a Station Black Out. This event would be applicable to most of the 70 plus NPP sites in the U.S., but as discussed in response to the SANDIA Uncertainty proposal, there are several sites where the wind and storm surge coming from a Level 5 hurricane would be at least as likely to cause an SBO as an earthquake 0.4 to 0.5g peak ground acceleration, pga for these sites at the 10^{-6} or 10^{-7}/yr probability of exceedence level. In a significant number of other NPP sites I believe such as is the case of the Surry NPP a 0.8 to 1.0g pga earthquake typical of a 10^{-6} to 10^{-7}/yr probability of exceedence level would more likely lead to an immediate gross loss of the containment leak tight integrity at penetrations resulting from differential settlement caused by foundation soil liquification or consolidation between the containment and adjacent safety related structures than would be caused by a longer term SBO.

I will also provide comments as requested by your email of 27 October 2010.

SOARCA Project Uncertainty Analysis
Karen Vierow
November 15, 2010

Similar to the SOARCA project, which was initially intended to evaluate a larger number of reactors than turned out to be feasible with the given time and resource restraints, the SOARCA Uncertainty Analysis effort must be justified as reasonable within the given programmatic restraints. Further, the sufficiency, or completeness, of the effort must be demonstrated.

A full level III PRA for each SOARCA scenario would require time and resources far beyond those which are available. The current effort will include one scenario for one reactor as a reasonable substitution approach. This approach has the merits of including establishment of the methodology for future Uncertainty Analyses and of allowing for depth of evaluation of the selected scenario. The definitions and detailed description of the methodologies in Section 2.4 are very helpful.

The case for completeness of the Peach Bottom Long Term Station Blackout Scenario must be made. That is, even though some low probability or low consequence aspects may be omitted, the high risk outcomes must be included.

The approach to not consider cross-correlation between uncertainty parameters needs to be justified or modified.

Some comments on the Oct. 19, 2010 Uncertainty Analysis draft report, which has been made available to the peer reviewers, follow.

- Several sections of the report describe the aspects of the Uncertainty Analysis effort but do not close the loop by relating these to the SOARCA Uncertainty Analysis objectives. Examples:
    - Section 2.3: the type of uncertainties targeted (aleatory and/or epistemic) should be stated up front.
    - Section 2.3: How this approach enables one to validate the Best Estimate SOARCA calculations should be explained.
- Section 2.1, item 5: consider rewording the "fewer failed MELCOR simulations" as this implied doubt about the robustness of the MELCOR code.
- Section 3.2, 1st sentence: This sentence seems essential to the report, however, this review is confused over the meaning. Please reword.
- Many sections of the report need better documentation of their justification, such as the selection of uncertainty parameters and/or their distributions. Under Recommendation 2, the solicitation of these data from "senior SNL and NRC technical staff…" is needed to clarify that the effort is less than exhaustive due to programmatic constraints. However, also needed is an explanation that the value of the current effort is not compromised.

SOARCA Uncertainty Analysis Plan

Uncertainty in estimation of health impact: Comments by Jacquelyn C. Yanch

The main health impact of the severe reactor accident is predicted to be latent cancer fatalities. The main source of uncertainty associated with estimating the number of latent cancer fatalities arises from taking risk estimates originally generated for a situation involving rapid radiation exposure of people (total dose delivered in ≤ 1 minute) and applying these risk estimates to the post-reactor-accident scenario where the total dose is delivered over a period of years to decades.

The SOARCA team plans to generate distributions to represent uncertainty in risk estimates in collaboration with Dr. Keith Eckerman. Their second (fallback) option [N. Bixler presentation, October 2010) is to use the spread of risk estimates generated via expert solicitation [NUREG 6555, 1997] to represent uncertainty. Since the experts in this solicitation were asked to supply estimates of risk of a large dose (1 Gy) delivered rapidly (1 min), a dose rate effectiveness factor (DREF) must be used in order to make the risk estimates applicable to the prolonged exposure situation. The SOARCA team uses a value of 2 for the DREF and plans to sample this value, for determination of uncertainty, from a distribution provided by the EPA [EPA 402-R-99-003, 1999]. In my opinion the distribution function provided by the EPA is not applicable to the situation involving very prolonged exposures and a broader range of DREF values should be included in the uncertainty estimation.

In reviewing available data, the NRCP reports that the DREF depends very significantly on the duration of the radiation exposure [NCRP 64, 1980]. Short-term exposures, such as those representative of typical occupational radiation scenarios, require a smaller value of DREF than long term exposures such as those lasting years or decades. Since the radiation exposure represented by returning home to elevated radiation levels following a severe reactor accident involves years or decades of exposure, a DREF appropriate to this scenario should be used. This value, according to NRCP 64, ranges from 6.6 to 12.8 with a best estimate of 10. However, the distribution function of DREF values provided by the EPA peaks at a value of 2 and the probability of sampling a value of 10, the most likely applicable value for the post-accident scenario, is vanishingly small. Thus, while the EPA distribution may be relevant to the rapid exposure scenario (eg. occupational radiation protection) it is not very relevant to the situation discussed here. This is echoed in BEIR V where it is stated that higher values of DREF reflect situations involving continuous irradiation until death. Therefore, in discussions with Dr. Eckerman, it is important that the more realistic DREF values be included in the estimate of uncertainty associated with latent cancer fatalities.

Uncertainty in the shape of the risk/dose model also exists. This uncertainty is accounted for in the current SOARCA approach of incorporating different threshold values below which no cancer fatalities would be observed.