SOARCA
Peer Review
DRAFT Comments

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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, Washington Safety Management Solutions;
John D. Stevenson, JD Stevenson Consulting Engineer Company;
Karen Vierow, Texas A&M University;
Jacquelyn C. Yanch, Massachusetts Institute of Technology

Prepared for:
Sandia National Laboratories
PO Box 5800 MS 1484
Albuquerque, NM 87185-1484

Signature Page

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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 was born out of 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 which 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 has been 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 represents a wealth of 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 may also be 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 the research institute 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 Bachelors 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, developed severe accident mitigation guidance, and 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 Washington 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. Dr. 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. Dr. 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. As of 2009 Professor Yanch also became a member of the MIT Department of Biological Engineering. Dr. 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. Finally, 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 is scheduled to submit the final version of this report in May 2010.

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 may also comment 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 Feb. 14, 2010 is the latest version available to the Committee at the time of preparation of this report.

The scope of the review does not include an Uncertainty Quantification and Sensitivity Analysis. Nor does it 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

Three 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 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 B.

The third and final 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, several 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 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 throughout 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 SOARCA Methods" the overall objective of SOARCA is to develop a body of knowledge regarding the realistic outcomes of severe reactor accidents. The corresponding and supporting objectives are summarized as follows:

- 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".

In overview, the SOARCA analysis and report has met its goals and objectives. Plant improvements and significant changes have been incorporated into the SOARCA models for the specific plants that are evaluated by SOARCA. The state-of-the-art severe accident modeling and behavior, has not only been implemented in SOARCA, but the state-of-the-art has actually been extended by the significant amount technical work and research developed and implemented in the study. In the area of severe accident communication, the technical community will benefit from the developments in SOARCA. The benefits and communication with other stakeholders beyond the technical community is beyond the scope of this review. The last objective, the quantification of offsite consequences was also met.

While the goals and objectives of SOARCA appear to be largely achieved, and in some cases the expectations actually exceeded, there are some observations worthy of note. The scope of this reviewer's comments are limited to the assigned topical area of accident sequence analysis. The individual observations are provided, in detail, in the next few paragraphs.

Consequence Analyses

One of the objectives of SOARCA is to develop current and realistic estimates of the potential site-specific offsite consequences from the more likely severe accidents for operating nuclear power plant. However, as is the case of all consequence analyses, SOARCA does indeed focus on only the most significant accident sequences. As such, the discussions of the impact of non-dominant or individually non-significant accident sequences in inevitable.

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

As stated previously, this discussion of "completeness" is typical issue with consequence analyses. That is, for consequence analyses it can be difficult to demonstrate completeness. The benefits to a frequency weighed 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 individual and collectively. It is realized that the frequency weighed approach can be both a benefit as well as a detriment. The detriment occurring where the results are misinterpreted, taken out of context, or manipulated without proper basis. However, this reviewer feels that 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 and report is developed by applying a method to two specific plants Surry and Peach Bottom. The application of the methods to two specific plants has both positive and negative aspects. The positive aspects are that with plant specific information, plant specific conclusions can be drawn and can be based on the specific design features, maintenance and operation practices at that particular site. The downside to this approach is that not all the plant specific features, both those features that reduce consequences as well as those that might increase consequences, are represented in the two plants chosen. As such, some conclusions are likely applicable to that site only and the results may not be typical.

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.

While an alternative to the current approach or analysis is not recommended or sought by this comment the results can be influenced in a material way by plant specific features.

Individual Accident Sequences

As part of the review of accident sequences in SOARCA the criteria used in the study were applied generically to various accident sequences in previously published PRA studies. The conclusion of this informal 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. The comparison does provide some assurance that the criteria was correctly applied at the same time the items 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. Both the successful operation as well as the failure modes under beyond design basis conditions are clearly significant in the analysis. The failure modes considered in the SOARCA analysis are, in the opinion of this reviewer likely however this illustrates another advantage of a frequency weighed approach where competing and important phenomena could be frequency weighed resulting in a more holistic view of risk and the key contributors.

Summary

The SOARCA analysis has met is 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. SOARCA is a state-of-the-art consequence analysis.

However, SOARCA is a consequence study and, as such, has issues associated with demonstrating completeness. Consequence studies are also limited in the ability to obtain the most utility from the final results. This is a result of the fact that they are difficult to change or modify 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 and the detriment of not reflecting the range of potential designs or the magnitude that 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 are not quantified and further limit the usefulness of the final result.

Evaluation by B. Clément

Summary

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

The SOARCA Project succeeded to achieve 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 programme on fission products behaviour 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 start-point. Besides, the foreseen methodology for uncertainty analysis is valid.

For work continuation, it is recommended: (i) to proceed in the future 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, (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 focussed on general documents describing the SOAECA objectives and methodology as well as on accident progression and source tem 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. This is discussed in the next section.

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

Accident Scenario Selection

SOARCA being 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 radio-nuclide 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 that is undoubtedly a state-of-the-art tool for core degradation but that not yet incorporates all the recent outcomes of researches 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 depressurisation of the RCS allows injection of water by the accumulators that delays the progression of the accident. Secondly, this avoids any high pressure melt ejection. In addition to this base case, scenarios with thermally-induced steam generator tube rupture were considered. Although the base case scenario is credible and corresponds to the best-estimate philosophy of SOARCA, uncertainties on different failure modes and locations must be taken into account.

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 if we are far or not from the σ criterion for flame acceleration and the λ criterion for detonation in order to evaluate . Those are given in the following document:

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 analysed 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 that 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 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 start-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 the method can allow to get a qualitative measurement of the impact of non modelled 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. A special 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 select a parameter's value falling 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 for work continuation

The objectives of the SOARCA Project were not to develop a full level 3 PRA. There is however a non deniable interest in developing level 2 and level 3 PRAs. Such actions, if possible, should be made in parallel with the continuation of SOARCA Project. Depending on the outcomes of new PRAs, it would be useful, or not, 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 US operating Nuclear Power Plants. Before deciding on an extension to the whole US fleet, it would be interesting to address other pilot plants representative of other designs such as BWRs with Mark 2 containment of 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 looked at with a deeper attention in the future.

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

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.

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

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

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 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 Team, several sensitivity analyses have provided a better understanding of the controlling phenomena and identified areas of 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, which was outside the scope of the SOARCA project.
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 four (4) volume 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 PRA) in order to more completely characterize the results and communicate risks.

Comments of Robert E. Henry

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. In this regard, there are two major comments that I believe need to be addressed. These are given below.

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

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

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

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

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.

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

In addition to restructuring the outline of the "best practices" document, there are some other features that should be included to document the manner in which the central estimates have been evaluated. These additional items are discussed below.

- The release fractions of the dominant chemical states provides the manner in which the fission products from the fuel become airborne in the core region. The transport of these fission products from the core, thorough the RCS and into containment, as well as their deposition in these regions is determined by the aerosol model. Typically, the aerosol densities within the reactor coolant system can be in excess of 100 grams per cubic meter, which is a very dense aerosol. Hence, the deposition within the RCS can be quite large and the manner in which this is calculated needs to be documented as part of the "best practices". I suggest that the benchmarks of the aerosol model with experiments such as the large scale ABCOVE tests, the DENONA test, etc., where available, be included in this "best practices" document. This is important to capture since the aerosol transport and deposition model is that feature of the calculation that determines the extent of airborne fission products in the containment that could be released to the environment. It is particularly key that this discussion be included, along with the benchmarks that are relevant to the aerosol densities typically encountered in the RCS and containment, to be assured that indeed a central estimate is justified.
- The accident progression within the core region from the intact fuel rods with "breakout" of molten zirconium that drains to the lower core region, eventual relocation of the molten debris from the core to the lower plenum and the controlling heat transfer to the RPV lower head need to be described. With the differences in geometry between the PWR and the BWR, as well as some potential differences between the U.S. commercial fleet PWR designs, for example upflow versus downflow core bypass, this should be

described in the "best practices" document since this will be consulted, evaluated and referenced in future studies. Furthermore, only through an understanding of this core melt progression and relocation to the lower plenum can the features that should be investigated through uncertainty analyses be clearly defined.

- The general public is well aware of the severe core damage accident that occurred in the Three Mile Island Unit 2 reactor. Any integral thermal-hydraulic model that is used to assess the timing for the onset of core damage, the release of fission products from the core, the extent of hydrogen generated in the core degradation, the transport of molten core debris to the lower plenum, etc. needs to be benchmarked with this accident. This benchmark evaluation needs to be either part of the SOARCA documentation or, at the minimum, referenced extensively in the other SOARCA reports. My preference would be the former, but I leave this to the judgment of the authors. In either case, the SOARCA reports should reference the insights/lessons learned from this benchmark and how this knowledge base is manifested in the analyses that are perform for the reference PWR in the SOARCA study.

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

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 was 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 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.

Peach Bottom SOARCA Document

No.	Peer Reviewer Comments	Response/Resolution	Peer Reviewer Evaluation of Response Resolution
1.	Why is siren used as 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 is 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 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 has 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.

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 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 non-evacuating 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 non-evacuating cohort. In some of the slowly developing sequences, 100% of the emergency phase risk is from non-evacuees.	If the non-evacuating public is properly informed, and elects not to follow the public officials recommendations to evacuate, they should be solely responsible for any negative consequences.

Individual Input on SOARCA Report

David Leaver

March 31, 2010

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 and a revised draft issued in February, 2010). There were also three 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 performed by SOARCA. The methodology to be used in the uncertainty analysis was discussed in the last peer review meeting and comments on this methodology were generated by the peer review team. I had not, however, seen the uncertainty results at the time of preparation of this individual input.

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 the Regulatory Information Conference in 2008 and 2009, previous NRC meetings with the ACRS as well as upcoming meetings where the draft documentation will have been made available to ACRS members, an extensive outside peer review (resulting in this peer review report), an upcoming public comment period and 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 are to be 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, and further indicated that because of their potential for risk, large seismic events should be assessed as part of a separate, future study which is to be 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. Based on a review of the Summary volume discussion in this area, my judgment is that none of these classes of accident events is likely to substantially alter the SOARCA findings on reactor risk. However, as indicated in the Summary volume, there would be benefits to applying more detailed best estimate, SOARCA-like methods to at least some of these classes of accident events. 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 SOARCA screening process will likely not be without controversy in the minds of some stakeholders, and further work on full-scope Level 3 may be beneficial for confirmatory purposes.

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 power community that the characterization of commercial LWR risk in previous studies was 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 B.5.b mitigative measures considered in SOARCA are in my view very important, partly because of the risk impact (though even without B.5.b measures the risks are predicted to be very small, zero early fatality risk and very low latent cancer risk), but also because of the fact that these measures provide margin for uncertainties in sequence selection and analysis, and make the 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 but a human reliability study that incorporates the measures into the SPAR models is scheduled to be released later this year.

- SOARCA objectives included updating earlier risk assessments, incorporating state-of-the-art analytical tools and insights from nearly three decades of research, and communication of severe accident-related aspects of reactor safety to stakeholders. While SOARCA was (properly so) performed with no agenda with regard to regulatory applications of the results, it would be appropriate, at some point after the final results are issued, to begin consideration of how the SOARCA methods and results could be used by licensees and in the regulatory process. Risk-informed regulations provide a framework for considering this, and the potential benefit is even better optimization of resources for assuring safety.

Review Objective

The object of this task was to perform an independent technical review of the approach and underlying assumptions and results obtained for the Peach Bottom and Surry SOARCA analyses. The 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, the review addressed by this document is focused on the selection and characterization of analyzed scenarios or sequences, and the treatment of mitigation measures and operator actions. Review comments are based on the SOARCA Project Report, Revision 1, dated February 15, 2010.

SOARCA Objective

As stated in the SOARCA Executive Summary, "[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."

It is 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 appeared to have been focused on this phenomenological modeling and were not used for the identification of sequences to be modeled or for the application of security-related mitigation improvements. These limitations which are discussed more fully below make it difficult to conclude that all the listed objectives were achieved. These limitations also appear to challenge the ability to effectively communicate severe-accident-related aspects of nuclear safety and to provide an update of NUREG/CR-2239.

In addition, the stated objectives also appear to be positively biased as indicated by the framing of the first supporting objective. This objective addresses "improvements and updates" as opposed to changes not reflected in earlier assessments. Although this reviewer agrees that there have been many improvements, there are also significant challenges associated with areas such as new fire model methods, increased on-line maintenance or generic issues such as the sump screen issue. A balance discussion should be included in the SOARCA report of the method used to select the changes for incorporation into this project in order to inform the reader as to potential limitations that may not have been addressed (see **Recommendation 1**).

Sequence Selection

As stated in the SOARCA Executive Summary, "SOARCA is intended to provide perspective on the likely (i.e., best estimate) outcomes of a severe accident at a nuclear power plant." A key challenge for the SOARCA project was the selection of the accident sequences, and therefore, it is not surprising that the report notes that "this was the subject of considerable deliberation, discussion, and review."

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." It also states 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 SOARCA report uses 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.

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 (see **Recommendation 2 and 3**).

The SOARCA Executive Summary shows that four accident sequences were selected for Surry's consequence analysis with three identified as external event related and one identified from the internal events PRA. The total frequency of these events is $2x10^{-5}$. Appendix B contains some variations to this list including an additional internal sequence associated with a spontaneous steam generator tube rupture (SGTR) (see **Recommendation 4**). A review of the internal event sequences contained in the Surry SPAR Model shows that the two internal event sequences selected for the SOARCA Project represent less than 15% of the internal events contribution to core damage and that depending on the approach used to bin the accident sequences several other sequences may have candidates for inclusion in the consequence analysis even if the 10^{-6} criterion was used (**See Recommendation 5 and 6**). Some of these sequences may be considered to have been bounded by the long-term and short-term station blackout (SBO) scenarios, but as currently written, these blackout scenarios appear to be addressing external event challenges and are separate from the internal event-related sequences.

Sequence Definition

In the SOARCA report, the terms "sequence" and "scenario" appear to be interchangeable. The ASME PRA Standard defines an accident sequence as "a representation in terms of an initiating event followed by a sequence of failures or successes of events (such as system, function, or operator performance) that can lead to undesired consequences, with a specified end state (e.g., core damage or large early release). In order to apply the stated screening criteria, it is important to clearly define the sequence structure, as sequences can be grouped by functions or can be subdivided to system trains and components. Subdivided sequences can easily have frequencies that are below the screening criteria. For example, the Surry SPAR Model has over 3,000 sequences with non greater than $1x10^{-6}$. These sequences could be easily grouped into a dozen sequence groups having similar characteristics. The types of sequences considered to be within scope and how and if they were combined is not clearly stated within the SOARCA report (**See Recommendation 7**).

Sequence Consistency and Frequencies

Not all of the sequences included within the SOARCA report appear to have been assigned frequencies. In addition, the approach of using the same frequency regardless of the presence or absence of mitigative action creates difficulty in understanding the connectivity between the sections within the report. This is complicated by variations in sequence descriptions and in the order used to discuss the sequences within various sections of the report. For example, the executive summary identifies the four selected scenarios and their associated frequencies for Surry. Appendix B, Section 3 identifies 13 scenarios and discusses the estimated frequency for a couple of the sequences while it does not discuss others. Section 7 analyzes 5 of these 13 scenarios. In addition, the executive summary identifies thermally-induced SGTR as a scenario while Section 3 treats this issue as one of several variants to the "Short-term SBO" scenario. To further add to the confusion, Section 7 includes a sequence frequency for an analyzed sequence (within the heading of each table) however, does not appear to differentiate between similar scenarios with the exception of TISGTR (**Recommendation 8 and 9**).

Treatment of Mitigation Measures and Operator Actions

A stated SOARCA objective is to evaluate the potential benefits of recent security-related mitigation improvements. The SOARCA Executive Summary Conclusion Section states that "all the identified scenarios could reasonably be mitigated." 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 concluding statement. It also results in incomplete frequency information as the frequencies of the sequences with the added actions are not determined. In addition, there did not appear to be any assessments performed as to the impact of earlier operator action failures on the addition of security-related actions. 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 (**See Recommendation 10**).

Conclusions

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

Recommendations

1. Provide a summary of the changes that are being incorporated in response to the first supporting objective. Consider rewording this objective to reflect a balanced consideration of risk-significant improvements and challenges.
2. Provide a better justification for the selected screening criteria.
3. Provide the risk profile that is being assumed for the assessment of each plant. Although it is understood that this profile is estimated and is developed based on multiple models, it is impossible to judge the degree of risk being capture by this analysis without a clear starting point.
4. Ensure that the presentation of accident sequences is consistent between the executive summary and the appendices.
5. Provide explicit mapping of the sequences from the set of initial sequences for those that met the screening criteria to those that were considered in the consequence analysis. Ensure that the frequency for each sequence is explicitly identified. Ensure that the reason for elimination of a sequence is clearly stated.
6. Account for all significant sequences.

7. Define the sequence framework that is being used in the SOARCA Project. Ensure that it is consistent with the screening criteria.
8. Provide a summary table within each appendix that identifies each sequence meeting the screening criteria, and its treatment within the accident progression and the emergency response sections. Give each sequence a unique identifier and address it in the same order within each section.
9. Include the identification and/or development of each sequence frequency within Section 3 of each appendix.
10. Performed a human reliability assessment for the identified security-related mitigation improvements or identify a conservative screening value so that all sequence frequencies can be calculated.

Individual Assessment by Kevin O'Kula to be inserted here.

Appendix A.1

J.D. Stevenson

Input to the SOARCA Peer Review Report

Dr. Stevenson's major area of input to the SOARCA study is extreme external hazard evaluation such as earthquake, tornado, precipitation and flooding applicable to a NPP site. Included in his evaluation is the response of structures, systems and components, SSC to all extreme or abnormal loads which could cause failure, damage or malfunction of Important to Safety SSC resulting in early release outside of containment of the reactor core radiological inventory. His input to the study is related to the prescribed limiting event (s) used in the study at either a 10^{-6}/yr probability to cause release of reactor core radioactive inventory release or 10^{-7}/yr probability of early containment failure and reactor core radiological inventory release occurrence or exceedence as compared to seismic induced long or short term station blackout.

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

The Surry site appears to be founded on a foundation media which is susceptible to liquification at the 10^{-6} or 10^{-7}/yr. probability of exceedence earthquake level as illustrated in Table 1 which is taken from a report prepared on the subject by M. Power. The report indicates the Surry NPP is adequate with significant margin against liquification at the 10^{-4}/yr probability of exceedence level. I have not seen a seismic hazard study for the Surry site, but based on a typical hazard curve taken from the ASCE 43-05 Standard shown in Figure 2, one would expect a slope factor of 2 for a factor of 10 decrease in a probability of exceedence level.

In Figure 3 is a summary of peak ground acceleration at median probability of exceedence at the 10^{-4}/yr level taken from a summary of median pga values developed during the NRC's IPEEE program. For the Surry site a value of about 0.24g pga for a mean, not median, 10^{-4}/yr probability of exceedence appears reasonable.

Applying the factor of 2 in going from a 10^{-4} to a 10^{-5} and another factor of 2 going from 10^{-5} to 10^{-6}/yr probability of exceedence level, a pga of 0.96g is obtained at the 10^{-6}/yr probability of exceedence level. As can be seen by comparing a 0.96g pga to the values given in Table 1, the site is expected to liquefy at the 10^{-6}/yr probability of exceedence earthquake level. There is seismic hazard data for the Surry site in NUREG 1488 and I am in the process of evaluating that data to see if it agrees with the 0.96pga value.

Given that the site would liquefy, it would be necessary to evaluate the effect of such liquification on the leak tightness of the containment. The following is Section 5 from the Power report.

"5. Consequences of Liquification

The estimated consequences of liquification in sand B and in the select fill, which are the susceptible soils underlying the critical structures of the auxiliary building and the control building, are settlements of the overlying structures due to post-earthquake dissipation of pore pressures in the liquefied soils. These reconsolidation settlements would tend to occur rather slowly after the earthquake, perhaps over a period of several hours or days. Based on data presented by Lee and Albaisa (1974) and Tokimatsu and Seed (1987), the magnitude of the reconsolidation settlements is estimated to be approximately 1 percent of the thickness of the layer of liquefied soil. This could lead to maximum total settlements of approximately 3 inches in the event of liquification of Layer B and 1 ½ inches in the event of liquification of the select fill. Differential settlement could occur across the building widths due to variations in the soil layer thickness. All of the total settlements could be differential with respect to adjunct non-settling Category 1 structures (reactor building and pile-supported fuel building). In addition to these reconsolidation settlements, some shear distortional differential settlements could occur within the select fill because that layer is the direct bearing support for the auxiliary building and control building. However, it is judged that such distortional settlements should be minor because of the dense nature of the fill and the thinness of the layer relative to the foundation width."

The report indicates a total settlement of 3.0 + 1.5 = 4.5 inches based on a 1.0 percent consolidation of the Type B soil layer and the engineered (select) backfill beneath the auxiliary and control buildings at the site.

There is no indication in the report of the relative conservatism of the one percent settlement of the liquefiable layers (i.e. best estimate mean, median, mean plus one or more standard deviations?

In my opinion, it is not obvious that every one of the typical 100+ penetrations in the containment could accommodate a 4.5 inch differential without significant leakage or rupture which might lead to early containment bypass.

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

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

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

As part of the SOARCA effort it has been determined that associated with the buildup of hydrogen levels in the containment to those needed for detonation, there would be a corresponding buildup of steam in the containment such that the inerting presence of steam in the containment atmosphere would preclude a hydrogen detonation.

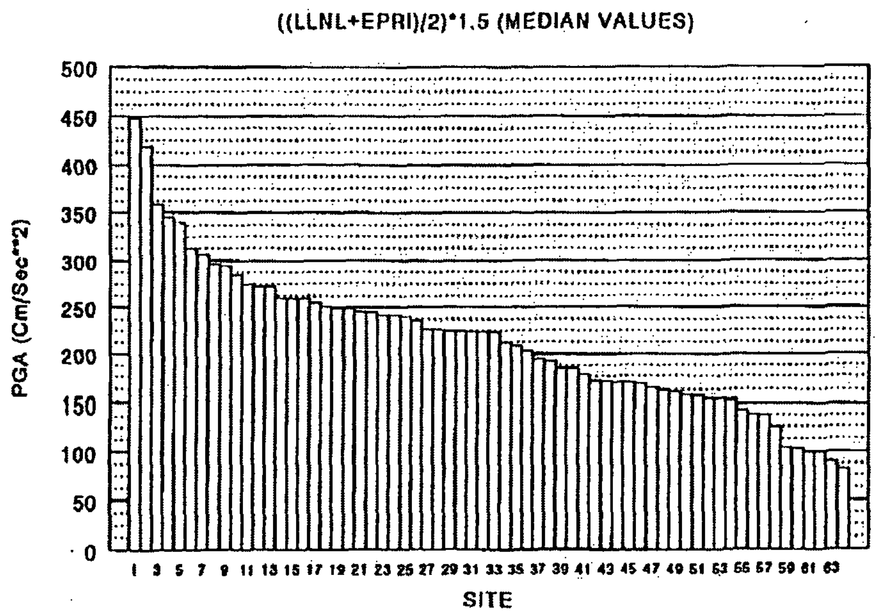
Table 1 Estimated Median Values of Free-Field Ground Surface Peak Accelerations Required to Cause Liquification at the Surry Site

	Median Acceleration to Cause Liquification (g)(3)			
	M5	M5.5	M6	M6.5
Free Field (Groundwater Level at El +5)				
Sand A	0.34 (± 20%)	0.31 (± 20%)	0.28 (± 20%)	0.25 (± 20%)
Sand B	0.40 (± 15%)	0.37 (± 15%)	0.34 (± 15%)	0.30 (± 15%)
Beneath Auxiliary Building And Control Building				
(a) Groundwater Level(1) at El -7				
Select Fill	>0.8 (>0.8)	>0.8 (0.75->0.8)	>0.8 (0.69->0.8)	>0.73 (0.60->0.8)
Sand B	0.40 (± 15%)	0.37 (± 15%)	0.34 (± 15%)	0.30 (± 15%)
(b) Groundwater Level at El +5				
Select Fill	0.78 (0.65->0.8)	0.72 (0.59->0.8)	0.65 (0.53-0.76)	0.56 (0.46-0.66)
Sand B	0.35 (± 15%)	0.32 (± 15%)	0.29 (± 15%)	0.26 (± 15%)

(1) The ground water level at -7 ft. assumes that mitigating dewatering pumps are active prior to the earthquake.
(2) Values in parentheses represent a possible range about the estimated accelerations due to uncertainties in the cyclic shear resistances of the soils.
(3) It should be assumed that the 10^{-6} or 10^{-7}/yr probability of exceedence earthquake hazard is earthquake magnitude 7.5 or above.

Figure 1 Tilting and settlement of a building in Niigata, Japan, as a result of soil liquefaction in the 1964 Niigata Earthquake.

8



PLANT SITE NUMBER

1	Pilgrim	33	Arkansas
2	Seabrook	34	Oyster
3	Sequoyah	35	Browns
4	Watts	36	Susquehanna
5	Yankee	37	Ginna
6	Maine	38	Surry
7	Oconee	39	Palisades
8	Indian	40	Davis
9	North	41	Calvert
10	Bellefonte	42	Fitzpatrick
11	Haddam	43	Shearon
12	Limerick	44	Nine
13	Clinton	45	Nine
14	Millstone	46	Fermi
15	Millstone	47	Byron
16	Millstone	48	Dresden
17	Summer	49	Monticello
18	Catawba	50	Braidwood
19	Lasalle	51	Perry
20	Peach	52	Hatch
21	Brunswick	53	Hatch
22	Mcguire	54	Point
23	Robinson	55	Kewaunee
24	Three	56	Wolf
25	Vogtle	57	Quad
26	Vermont	58	Prairie
27	Hope	59	Grand
28	Salem	60	Farley
29	Arkansas	61	River
30	Zion	62	Big
31	Beaver	63	Crystal
32	Beaver	64	Comanche

Figure 2 Median Peak Ground Acceleration at the 10^{-4}/yr Probability of Exceedence Level

Note: Surry NPP is Plant No. 38 with a median pga of 0.18g. To obtain the mean value a multiplication factor of 1.25 has been used to determine the mean value.

a.

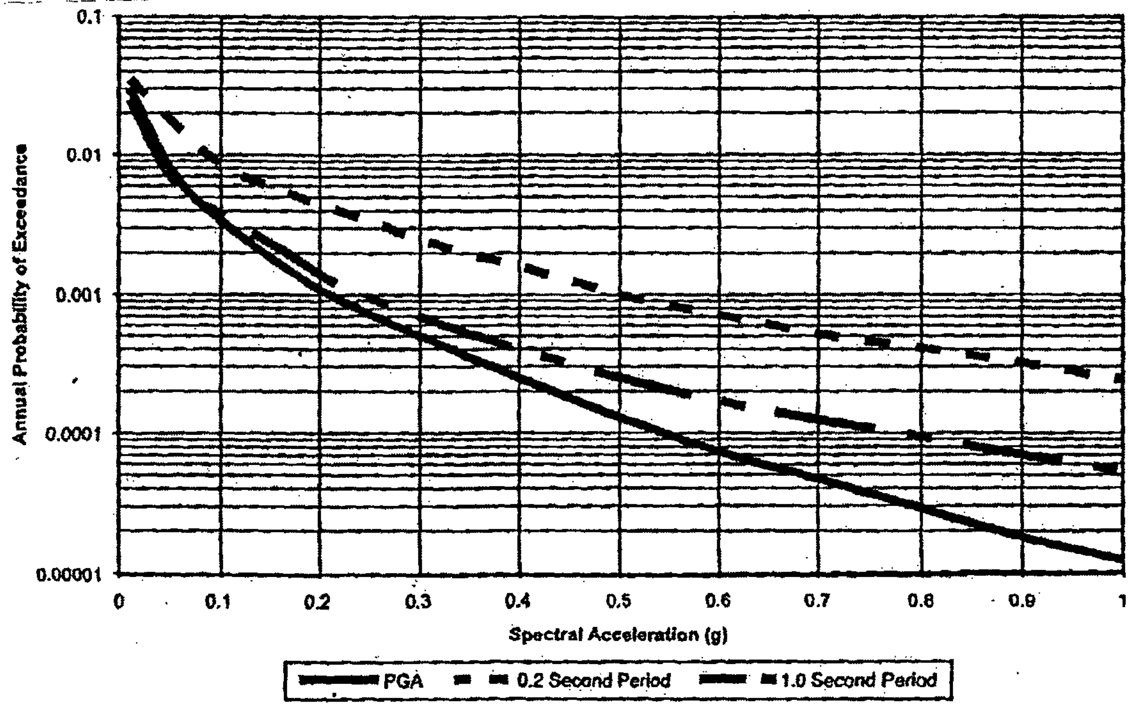


Figure 3 Typical Hazard Curve Taken From Figure 2-1 of ASCE Standard 43-05

ATTACHMENT A

APPENDIX F

Soils Liquefaction Analysis for Surry

by

Maurice S. Power
Principal Engineer

GeoMatrix Consultants, Inc.

TABLE OF CONTENTS

1. INTRODUCTION

This report describes a liquefaction fragility assessment conducted for the Surry nuclear power plant, Virginia. The specific objectives of the study are to estimate median values of free field ground surface peak acceleration required to cause liquefaction at the site and the associated consequences of liquefaction. It is our understanding that the critical structures at the site are the reactor building, control building, and auxiliary building. Therefore, our assessments have focused on liquefaction potential beneath these structures as well as in the free field. The results of this study will be used in a probabilistic risk assessment (PRA) of the plant.

A number of documents furnished by Sandia National Laboratories and by EQE Incorporated have been reviewed and utilized in conducting this study. These documents included the following:

1. Surry Plant Final Safety Analysis Report (FSAR), Section 2.4 Geology, dated 12-1-69, and Section 2.5 Seismology, dated 12-1-69 and 2-13-70.

2. Surry Plant Preliminary Safety Analysis Report (PSAR), Supplement S9.12, pp. S9.12-1 to S9.12-6, dated 11-15-67; Appendix S9.12A, pp. S9.12 A-1 to S9.12A-8, dated 12-5-67; Appendix S9.12B, pp. S9-12B-1 and S9.12B-2 dated 11-16-67, Appendix 9.12C, pp. S9.12C-1 to S9.12C-5, dated 11-15-67, Table S9.12C-1, and Figure S9.12C-1 dated 11-22-67; Appendix S9.12D, pp. S9.12D-1 to S9.12D-6 dated 11-24-67 and Figures S9.12D-1 to S9.12D-3 dated 11-22-67.

3. Surry Plant PSAR, Amendment 5, dated 12-7-67.

4. Dames and Moore report dated November 17, 1967, "Report Environmental Studies, Proposed Power Plant, Surry, Virginia, Virginia Electric and Power Company."

5. R.V. Whitman report dated 8-11-67 to Stone & Webster Engineers on Foundation Dynamics

6. EQE Incorporated letter of July 28, 1988 to M.S. Power, Geomatrix Consultants re: Median peak accelerations, base shear forces, and static bearing pressures for structures included in the Probabilistic Risk Analysis performed by EQE.

7. EQE Incorporated letter of August 16, 1988 to M.S. Power, Geomatrix Consultants re: Base shear stresses for structures included in the Probabilistic Risk Analysis performed by EQE.

2. SUBSURFACE CONDITIONS

The plan arrangement of the nuclear power plant complex is shown in Figure 1. Cross sections that show the facilities in relation to the subsurface soil conditions are presented in Figures 2 and 3. The foundation soils of interest for this study are:

<u>Sand A:</u> The layer typically exists between elevations 0 and -10 feet (26.5 to 36.5 feet below the plant finished grade). The layer does not underlie the critical structures. It was included in the analysis for completeness because its liquefaction potential had been addressed in the PSAR and FSAR.

<u>Sand B:</u> The layer typically exists between elevations -20 and -40 feet (46.5 to 66.5 feet below plant finished grade). It underlies the auxiliary building and the control building (both founded at elevation -2 feet) at depth, but the reactor building extends below the layer.

<u>Sand C:</u> Sand C is found at approximately elevation -58 feet on the average (approximately 85 feet below plant finished grade). The layer is typically interlensed with clay and the cumulative thickness of sand lenses is typically 5 feet or less. Sand C (where present) is approximately 18 feet below the mat foundation of the reactor building (at elevation -40).

<u>Select fill:</u> Beneath the auxiliary building and the control building, as well as beneath the fuel building, Sand A was excavated and replaced with select granular fill. The fill was reported in the FSAR to be compacted to a density equal to or exceeding 95 percent of the maximum density obtained using ASTM compaction test method 1557-66. The select fill provides direct bearing support for the mat foundations of the auxiliary building and the control building.

Groundwater levels were reported in the FSAR to be at elevation +5 feet in the free field. A permanent dewatering system was installed around the perimeter of the reactor buildings. The dewatering system is reported (FSAR) to maintain piezometric levels at or below elevation -30 feet beneath the reactor building and at or below elevation -7 feet beneath the auxiliary building and control building. In liquefaction potential evaluations originally carried out for the plant (PSAR and FSAR), the aforementioned piezometric levels were assumed; however, for the auxiliary building and control building, analyses were also carried out for a piezometric level of +5 feet to cover the possibility of the drainage system ceasing to depress the piezometric head in Sand B.

3. ASSESSMENT OF LIQUEFACTION RESISTANCE OF SOILS

Assessment of free field peak ground accelerations required to cause liquefaction requires two evaluations: (1) an evaluation of the cyclic shear stress, τ_L, or the cyclic stress ratio, $(\tau/\bar{\sigma})_L$ (where $\bar{\sigma}$ is the pre-earthquake effective vertical stress), required to cause liquefaction of the soils; and (2) an evaluation of the earthquake-induced cyclic shear stress or stress ratio $(\tau/\bar{\sigma})_E$ as a function of the free field peak acceleration at the ground surface. From these two evaluations, the acceleration levels causing the induced stresses or stress ratios to equal those causing liquefaction are obtained. The assessment of the cyclic stress ratios required to cause liquefaction is summarized in sections 3.1 through 3.3. Section 4 summarizes the assessment of the stress ratios induced by the earthquake ground shaking and the corresponding acceleration levels causing liquefaction.

3.1 Liquefaction Resistance of Sand A and Sand B

The present state of practice of evaluating the liquefaction potential of insitu soil layers generally relies on insitu measurements of the resistance of the soils to a penetration device and empirical correlations relating the penetration resistance to the cyclic stress ratio required to cause liquefaction. Typically, the resistance measure is the number of blows per foot required to drive a standard sampler into the soil at the base of a drill hole (Standard Penetration Test, SPT). The resistance to penetration of a static cone penetrometer (Cone Penetrometer Test, CPT) is also often used as a resistance measure.

At the Surry plant site, there are a number of SPT results in Sands A and B. These were used to assess the liquefaction resistance of these soil layers. The empirical correlation that was used to relate the normalized SPT penetration resistance, N_1 (i.e. the penetration resistance adjusted to a common effective vertical stress of 2 ksf), to the cyclic stress ratio causing liquefaction is the widely used correlation developed by Seed and his coworkers. The current version of this correlation for a magnitude 7-1/2 earthquake is shown in Figure 4 (Seed and others, 1985). As shown, the cyclic stress ratio causing liquefaction for a given magnitude earthquake is a function of the percentage of silty and clayey fines in the sand as well as the penetration resistance. Factors are presented by Seed and others (1985) to adjust the ordinates of the curves in Figure 4 to magnitudes other than 7-1/2. The factors result in increasing values of $(\tau/\bar{\sigma})_L$ with decreasing magnitudes.

One other adjustment should be made to the values of cyclic stress ratio obtained from Figure 4. It has been found that these stress ratios decrease somewhat with increasing effective vertical stress, $\bar{\sigma}$, and the values in Figure 4 are applicable to $\bar{\sigma} = 2$ ksf. A relationship recently developed by Seed and his coworkers (Seed, 1988, personal communication) was used to make this adjustment.

The normalized penetration resistance values obtained from SPT tests in the plant site borings (summarized in the FSAR) are shown in Figures 5 and 6 for Sands A and B, respectively. In obtaining these plots, the blow counts have not only been normalized to an effective overburden pressure of 2 ksf (using the chart presented by Seed and others, 1985), they have also been adjusted to those of a clean sand (i.e. sand with ≤ 5 percent fines) using the relative position of the curves in Figure 4 along with data presented in the FSAR describing the fines contents of the sands. These data indicate that the fines content of Sand A and Sand B are typically equal to or greater than 10 percent and 25 percent, respectively. Based on Figure 4, an upward N_1 adjustment of 2 blows/foot for Sand A and 5 to 7 blows/foot for Sand B (depending on the unadjusted N_1 value) was made to adjust the N_1 values to those of a clean sand.

In assessing the cyclic stress ratios causing liquefaction in Sands A and B, representative or characteristic blow counts for the layers must be selected from the scattergrams in Figures 5 and 6. Seed (personal communication, 1984 and 1988) indicated that a characteristic blow count that is consistent with how the empirical correlation was developed is the 33rd percentile blow count of the distribution after eliminating obvious outliers. Accordingly, the N_1 values selected for Sands A and B from the plots in Figures 5 and 6 are equal to 15 and 18, respectively. Using these N_1 values, the curve for clean sand in Figure 4, and appropriate adjustment factors for earthquake magnitude and effective vertical stress, values of cyclic stress ratio causing liquefaction in Sand A and in Sand B were obtained.

Seed and others (1985) describe the sensitivity of N_1 values to the exact techniques used in conducting Standard Penetration Tests. In fact, the designation $(N_1)_{60}$ in Figure 4 refers to a specific type of drophammer used for the SPT that delivers on the average 60 percent of the theoretical free-fall energy to the rods to which the sampler is attached. Since the details of the techniques used in conducting SPT tests at the site are not known, there are some uncertainties in the cyclic stress ratios causing liquefaction.

The influence of these uncertainties on values of peak ground acceleration causing liquefaction is discussed in Section 4.

The PSAR and FSAR contain dynamic (cyclic) test results on undisturbed samples of sand from layers A and B and an evaluation of the cyclic stress ratios causing liquefaction using these test results. The test results are few and widely scattered. Experience since the late 1960's when these facts were made has demonstrated the extreme difficulty in obtaining cyclic test results representative of insitu conditions, which has, in turn, spurred the development and utilization of empirical correlations and insitu test data in characterizing liquefaction resistance, as summarized above. Nevertheless, previous cyclic test results and interpretations were reviewed during the present study. It was found that when the cyclic test results were interpreted using correction factors established in later years, the cyclic stress ratios causing liquefaction interpreted from these tests are in good agreement with those interpreted during this study from the empirical correlations and insitu test data.

3.2 <u>Liquefaction Resistance of Sand C</u>

There are virtually no insitu test data nor laboratory test data in Sand C due in part to the lenticular nature of the deposit and its slight thickness (equal to or less than 5 feet thick). Based on the fact that the layer is relatively old geologically (of Miocene age, whereas the overlying Sands A and B are of Pleistocene age) and thin, it is judged that this layer has a high resistance to liquefaction and does not pose a significant hazard to the plant structures.

3.3 <u>Liquefaction Resistance of Select Fill</u>

Based on the minimum degree of compaction requirement for the fill stated in the FSAR, it is judged that the relative density of the fill should be approximately equal to or greater than 80 percent. The cyclic shear resistance of the fill was estimated using published laboratory cyclic test results for granular soils compacted to various relative densities (Seed, 1979; Lee and Seed, 1967) along with consideration of the beneficial effect of aging of the fill since placement (Seed, 1979). In addition, the liquefaction resistance of the fill was estimated on the basis of an assumed N_1 value for the fill; for a well compacted granular fill, it is judged that N_1 should be approximately 25 blows/foot or higher. The effect of possible variations in the liquefaction resistance of the fill on the acceleration levels to cause liquefaction is discussed in the following section.

4. ASSESSMENT OF EARTHQUAKE-INDUCED STRESS RATIOS AND PEAK GROUND ACCELERATIONS CAUSING LIQUEFACTION

For free-field conditions, the ratio of the earthquake induced cyclic shear stress to the pre-earthquake effective vertical stress, $(\tau/\bar{\sigma})_E$, can be obtained using the widely used simplified procedure (Seed and Idriss, 1971; Seed and others, 1983):

$$\left(\frac{\tau}{\bar{\sigma}}\right)_E = a \cdot \frac{\sigma}{\bar{\sigma}} \cdot r_d \cdot 0.65 \qquad (1)$$

where a - peak acceleration at the ground surface in the free field

σ - total vertical stress at a depth of interest below the ground surface

$\bar{\sigma}$ - effective vertical stress at the same depth

r_d - depth-dependant shear stress reduction factor (mainly accounting for the reduction of peak ground acceleration with depth below the ground surface)

0.65 - factor to obtain average shear stress from peak shear stress

By equating the earthquake-induced stress ratio, $(\tau/\bar{\sigma})_E$, to the stress ratio required to cause liquefaction, $(\tau/\bar{\sigma})_L$, the peak ground acceleration, a, causing liquefaction is obtained.

For conditions beneath structures, a modified form of Equation (1) was used to incorporate the shear stresses induced in the soil by the structures' response to the earthquake ground motions:

$$\left(\frac{\tau}{\bar{\sigma}}\right)_E = \left[\frac{\tau_b}{\bar{\sigma}} + \frac{a_s \cdot r_d \cdot \sigma_s}{\bar{\sigma}}\right] \cdot 0.65 \qquad (2)$$

where τ_b = shear stress-induced in the soil at a depth of interest below the structure due to the structure's base shear stress, τ_s, at the foundation-soil interface.

a_s = peak acceleration at the base of the structure.

σ_s = component of the total vertical stress due to the soil weight between the base of the structure and the depth of interest ($\sigma_s = \gamma_t \cdot z$ where γ_t is the total unit weight of soil and z is the depth below the base of the structure).

and other parameters are as defined previously.

In essence, the first term on the right hand side of Equation 2 represents the shear stress induced in the soil layer due to base shear transmitted by the responding structures and the second term represents the shear stress induced in the soil layer by the inertial response of the soils beneath the structure.

Values for the base shear stress, τ_s, transmitted by the structures and the acceleration at the base of the structures, a_s, as a function of the free-field ground surface acceleration, a, were provided by EQE from their soil-structure interaction (SSI) analyses carried out for the PRA. In the SSI analyses, embedment effects (if any) were neglected for the auxiliary building and the control building, which may be conservative. The shear stress, τ_b, induced at some depth beneath the structure due to the structures' base shear was estimated using elastic, static shear stress influence factors.

In evaluating the vertical effective stress, $\bar{\sigma}$, elastic solutions were also used to obtain the stress distribution with depth resulting from the structures' bearing pressures. Bearing pressures were provided by EQE. The variation of r_d with depth below the structures was assumed to be the same as the variation with depth below the ground surface in the free field (i.e. structure-foundation soil interface taken as zero depth).

Using Equation 2, values of the induced cyclic stress ratio, $(\tau/\bar{\sigma})_E$, were obtained as a function of free-field peak ground surface acceleration, a. (The relationships between $(\tau/\bar{\sigma})_E$ and a are nonlinear because of nonlinear relationships between a and a_s, and a and τ_s obtained in the SSI analyses by EQE.) Values of a causing liquefaction were then obtained by equating $(\tau/\bar{\sigma})_E$ with the cyclic stress ratio required to cause liquefaction, $(\tau/\bar{\sigma})_L$. Because Equation 2 involves greater uncertainty in the estimates than those obtained using the free-field formulation of Equation 1, the results were interpreted somewhat conservatively.

Table 1 provides a summary of the free-field ground surface peak accelerations causing liquefaction obtained from the analyses. Estimated peak accelerations causing liquefaction are summarized for four earthquake magnitudes (5, 5.5, 6, and 6.5) for Sands A and B in the free-field and for the select fill and Sand B beneath the auxiliary building and control building. Consistent with prior analyses presented in the FSAR, peak accelerations are presented for two piezometric levels in the soils below the auxiliary building and the control room — elevation -7, which is the expected highest piezometric level beneath these structures due to the influence of the permanent dewatering system; and elevation +5, which is the level that would exist beneath the structures if the dewatering system were not draining the soils beneath the structures as expected. (The latter water level would thus appear to represent an unlikely condition.) Analyses are not presented for Layer C because, as previously noted, it is judged that this layer is very resistant to liquefaction and any consequences of liquefaction in the layer would be insignificant. The SSI results for the reactor building obtained by EQE are also indicative of very low shear stresses induced in Sand C by the reactor building.

Possible ranges in the estimated values of peak ground acceleration causing liquefaction due to uncertainties in the cyclic shear resistances of the soils are summarized in the entries in parentheses in Table 1. For natural Sands A and B, the ranges reflect our judgment as to a possible range of N_1 values due to unknown details of conducting the Standard Penetration Tests at the site. Considering the geologic age of these sands, it is also our judgment that values in the upper half of the ranges are more likely than values in the lower half. For select fill, the ranges in the table reflect our judgment as to a possible range of relative densities to which the fill was compacted (given that it was compacted to the compaction standard stated in the FSAR) or corresponding range of N_1 values.

The peak accelerations summarized in Table 1 are median (50th percentile) values because the correlation for liquefaction resistance (Figure 4) has been interpreted by its developer as a median curve (Seed, 1988, personal communication) and the estimates of induced stress ratios are also considered to be median estimates. In a previous study (Power and others, 1986), a probabilistic distribution was developed for the liquefaction resistance curves. Development of the distribution involved quantification of the expert judgment of the developer of the correlation, Professor H.B. Seed. However, since that work was done, data have been added and reinterpreted and the correlation has been revised. With regard to the current correlation (Figure 4), Professor Seed's preliminary judgment (Seed, 1988, personal communication) is that the band of uncertainty about the median line has narrowed such that the 5th and 95th percentiles of the distribution for $(\tau/\bar{\sigma})_L$ may vary by a factor of only about 1.15 to 1.2 from the median curve. Liao and others (1988) recently quantified the uncertainty in the cyclic stress ratio causing liquefaction; however, the correlation they derived is different from the correlation in widespread general use that is shown in Figure 4.

The foregoing observations suggest that, for purposes of the present PRA, uncertainty in the liquefaction correlation could be included as summarized above. It could be assumed that the variation of peak accelerations about median values is about the same as the variation in the liquefaction resistance, i.e., a variation by a factor of 1.15 to 1.2 from median values at the 5th and 95th percentile levels. A log-normal distribution could reasonably be used to model the uncertainty. The uncertainty could be increased to incorporate uncertainty in the induced stress ratios. It is judged that this would increase the overall uncertainty to a factor of about 1.25 at the 5th and 95th percentile levels. In addition to the variation about median values, uncertainty in the median values, as discussed previously and summarized in Table 1 due to uncertainty in the N_1 values or relative density of the soil, could be included.

5. CONSEQUENCES OF LIQUEFACTION

The estimated consequences of liquefaction in Sand B and in the select fill, which are the susceptible soils underlying the critical structures of the auxiliary building and the control building, are settlements of the overlying structures due to post-earthquake dissipation of pore pressures in the liquefied soils. These reconsolidation settlements would tend to occur rather slowly after the earthquake, perhaps over a period of several hours or days. Based on data presented by Lee and Albaisa (1974) and Tokimatsu and Seed (1987), the magnitude of the reconsolidation settlements is estimated to be approximately 1 percent of the thickness of the layer of liquefied soil. This could lead to maximum total settlements of approximately 3 inches in the event of liquefaction of Layer B and 1½ inches in the event of liquefaction of the select fill. Differential settlement could occur across the building widths due to variations in the soil layer thicknesses. All of the total settlements could be differential with respect to adjacent non-settling Category 1 structures (reactor building and pile-supported fuel building). In addition to these reconsolidation settlements, some shear distortional differential settlements could occur within the select fill because that layer is the direct bearing support for the auxiliary building and control building. However, it is judged that such distortional settlements should be minor because of the dense nature of the fill and the thinness of the layer relative to the foundation width.

An assessment was also made of the potential for lateral movements of the structures toward the slope of the discharge canal (Figure 1) in the event of liquefaction. Simplified Newmark-type procedures as presented by Makdisi and Seed (1978) were utilized in estimating the deformations. It was assumed that the water level elevation in the canal was approximately equal to the ground water elevation. Based on these analyses, it is judged that lateral movements of the structures would be small (less than 1 inch) for levels of peak ground acceleration equal to or less than 1.5 times the accelerations required to cause liquefaction.

REFERENCES

Liao, S.C., D. Veneziano, and R.V. Whitman,(1988) "Regression Models for Evaluating Liquefaction Probability", Journal of Geotechnical Engineering, ASCE, 111 (4): 389-411.

Lee, K.L., and A. Albaisa (1974) "Earthquake Induced Settlement in Saturated Sands", Journal of the Soil Mechanics and Foundations Division ASCE 100(GT4): 387-406.

Makdisi, F. and Seed, H.B. (1987) "Simplified Procedure for Estimating Dam and Embankment Earthquake-induced Deformations", Journal of the Geotechnical Engineering Division, ASCE 104 (GT7): 849-867.

Power, M.S., V. Berger, R.R. Youngs, K.J. Coppersmith, and D.W. Streiff (1986) "Evaluation of Liquefaction opportunity and liquefaction Potential in the San Diego, California Urban Area", Report submitted by Woodward-Clyde Consultants to the U.S. Geological Survey under contract 14-08-0001-20607.

Seed, H.B. (1979) "Soil Liquefaction and Cyclic Mobility Evaluation for Level Ground During Earthquakes," Journal of the Geotechnical Engineering Division, ASCE 105(GT2): 201-255.

Seed, H.B., and I.M. Idriss (1971) "Simplified Procedure for Evaluating Soil Liquefaction Potential," Journal of the Soil Mechanics and Foundations Division, ASCE 97(SM9): 1249-1273.

Seed, H.B., I.M. Idriss, and I. Arango (1983) "Evaluation of Liquefaction Potential Using Field Performance Data," Journal of Geotechnical Engineering, ASCE 109(3): 458-482.

Seed, H.B., K. Tokimatsu, L.F. Harder, and R.M. Chung (1985) "Influence of SPT procedures in Soil Liquefaction Resistance Evaluations", Journal of Geotechnical Engineering, ASCE, 111 (12): 1425-1445.

Tokimatsu, K. and Seed, H.B. (1978) "Evaluation of Settlements in Sands due to Earthquake Shaking ", Journal of Geotechnical Engineering, ASCE 113 (8): 861-878.

REFERENCES

Liao, S.C., D. Veneziano, and R.V. Whitman,(1988) "Regression Models for Evaluating Liquefaction Probability", Journal of Geotechnical Engineering, ASCE, 111 (4): 389-411.

Lee, K.L., and A. Albaisa (1974) "Earthquake Induced Settlement in Saturated Sands", Journal of the Soil Mechanics and Foundations Division ASCE 100(GT4): 387-406.

Makdisi, F. and Seed, H.B. (1987) "Simplified Procedure for Estimating Dam and Embankment Earthquake-induced Deformations", Journal of the Geotechnical Engineering Division, ASCE 104 (GT7): 849-867.

Power, M.S., V. Berger, R.R. Youngs, K.J. Coppersmith, and D.W. Streiff (1986) "Evaluation of Liquefaction opportunity and liquefaction Potential in the San Diego, California Urban Area", Report submitted by Woodward-Clyde Consultants to the U.S. Geological Survey under contract 14-08-0001-20607.

Seed, H.B. (1979) "Soil Liquefaction and Cyclic Mobility Evaluation for Level Ground During Earthquakes," Journal of the Geotechnical Engineering Division, ASCE 105(GT2): 201-255.

Seed, H.B., and I.M. Idriss (1971) "Simplified Procedure for Evaluating Soil Liquefaction Potential," Journal of the Soil Mechanics and Foundations Division, ASCE 97(SM9): 1249-1273.

Seed, H.B., I.M. Idriss, and I. Arango (1983) "Evaluation of Liquefaction Potential Using Field Performance Data," Journal of Geotechnical Engineering, ASCE 109(3): 458-482.

Seed, H.B., K. Tokimatsu, L.F. Harder, and R.M. Chung (1985) "Influence of SPT procedures in Soil Liquefaction Resistance Evaluations", Journal of Geotechnical Engineering, ASCE, 111 (12): 1425-1445.

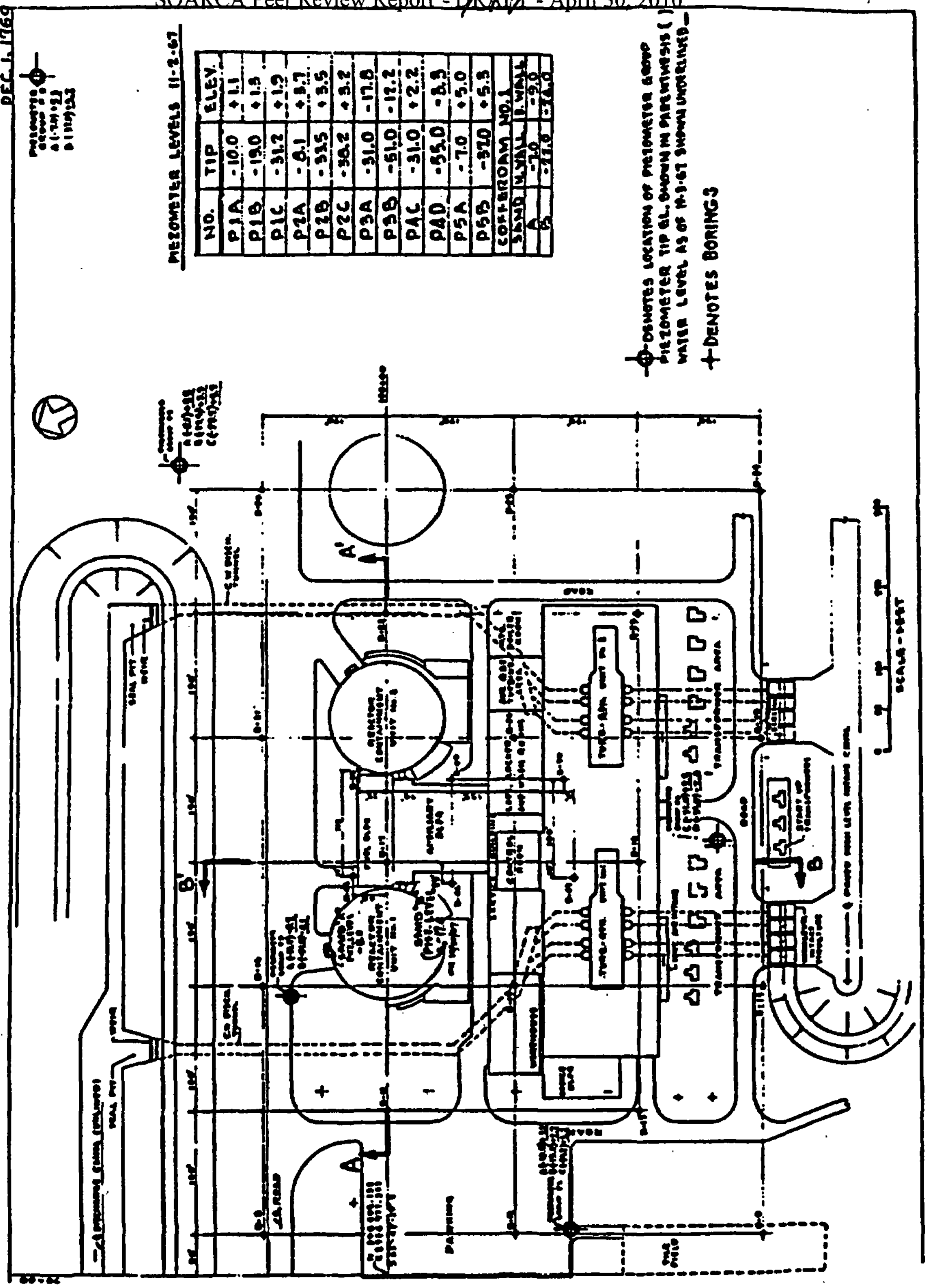
Tokimatsu, K. and Seed, H.B. (1978) "Evaluation of Settlements in Sands due to Earthquake Shaking ", Journal of Geotechnical Engineering, ASCE 113 (8): 861-878.

TABLE 1

ESTIMATED MEDIAN VALUES OF FREE-FIELD GROUND SURFACE PEAK ACCELERATIONS REQUIRED TO CAUSE LIQUEFACTION

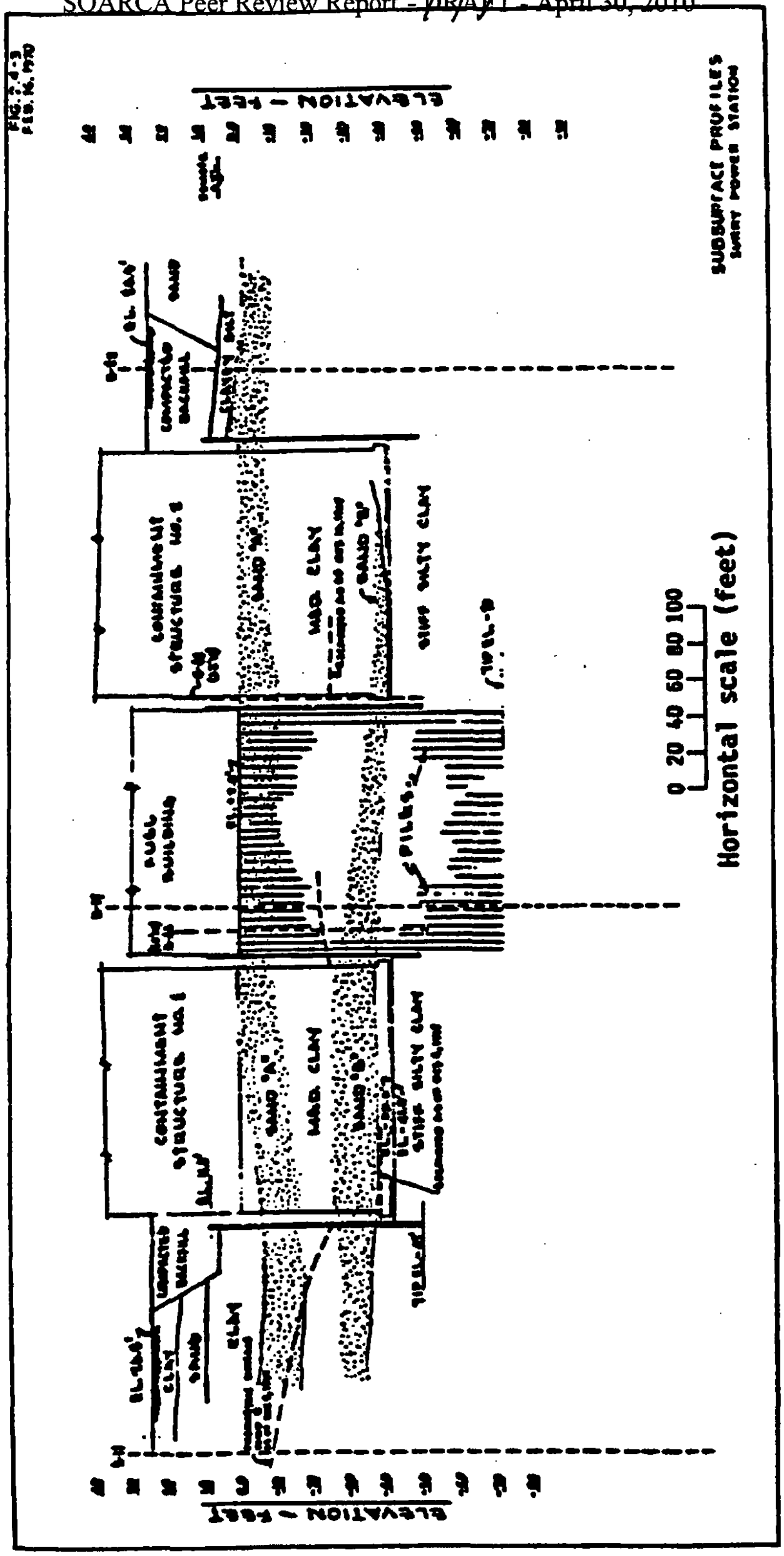
	Median Acceleration to Cause Liquefaction (g)			
	M5	M5.5	M6	M6.5
Free field (Groundwater Level at El +5)				
Sand A	0.34 (± 20%)	0.31 (± 20%)	0.28 (± 20%)	0.25 (± 20%)
Sand B	0.40 (± 15%)	0.37 (± 15%)	0.34 (± 15%)	0.30 (± 15%)
Beneath Auxiliary Building and Control Building				
(a) Groundwater Level at El -7				
Select Fill	>0.8 (>0.8)	>0.8 (0.75->0.8)	>0.8 (0.69->0.8)	>0.73 (0.60->0.8)
Sand B	0.40 (± 15%)	0.37 (± 15%)	0.34 (± 15%)	0.30 (± 15%)
(b) Groundwater Level at El +5				
Select Fill	0.78 (0.65->0.8)	0.72 (0.59->0.8)	0.65 (0.53-0.76)	0.56 (0.46-0.66)
Sand B	0.35 (± 15%)	0.32 (± 15%)	0.29 (± 15%)	0.26 (± 15%)

Note: Values in parentheses represent a possible range about the estimated accelerations due to uncertainties in the cyclic shear resistances of the soils.



GEOMATRIX	PLAN OF POWER PLANT From FSAR	Figure 1 Project No. 1134A

F-17

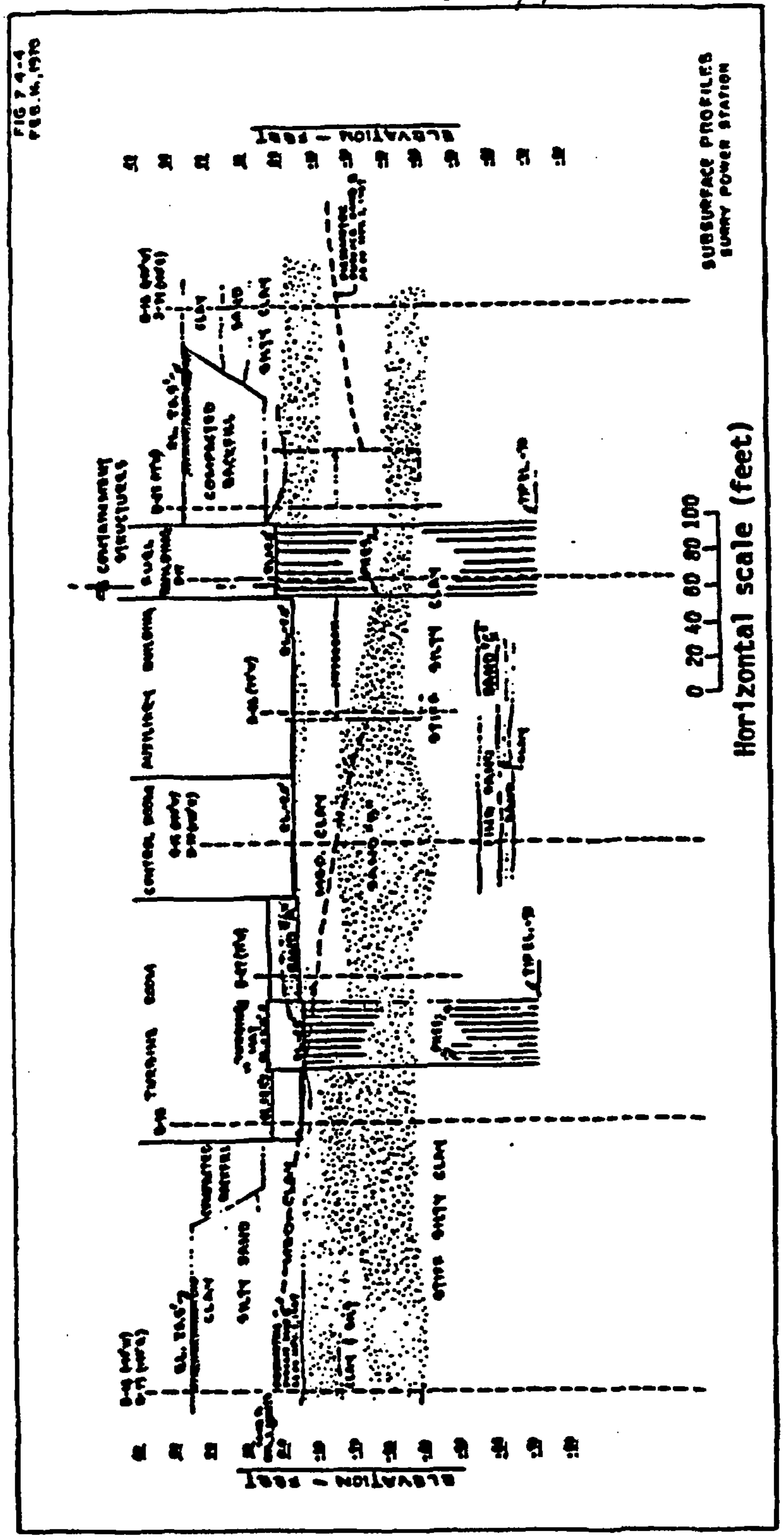


GEOMATRIX

SECTION A-A'
From FSAR

Figure
2

Project No.
1134A



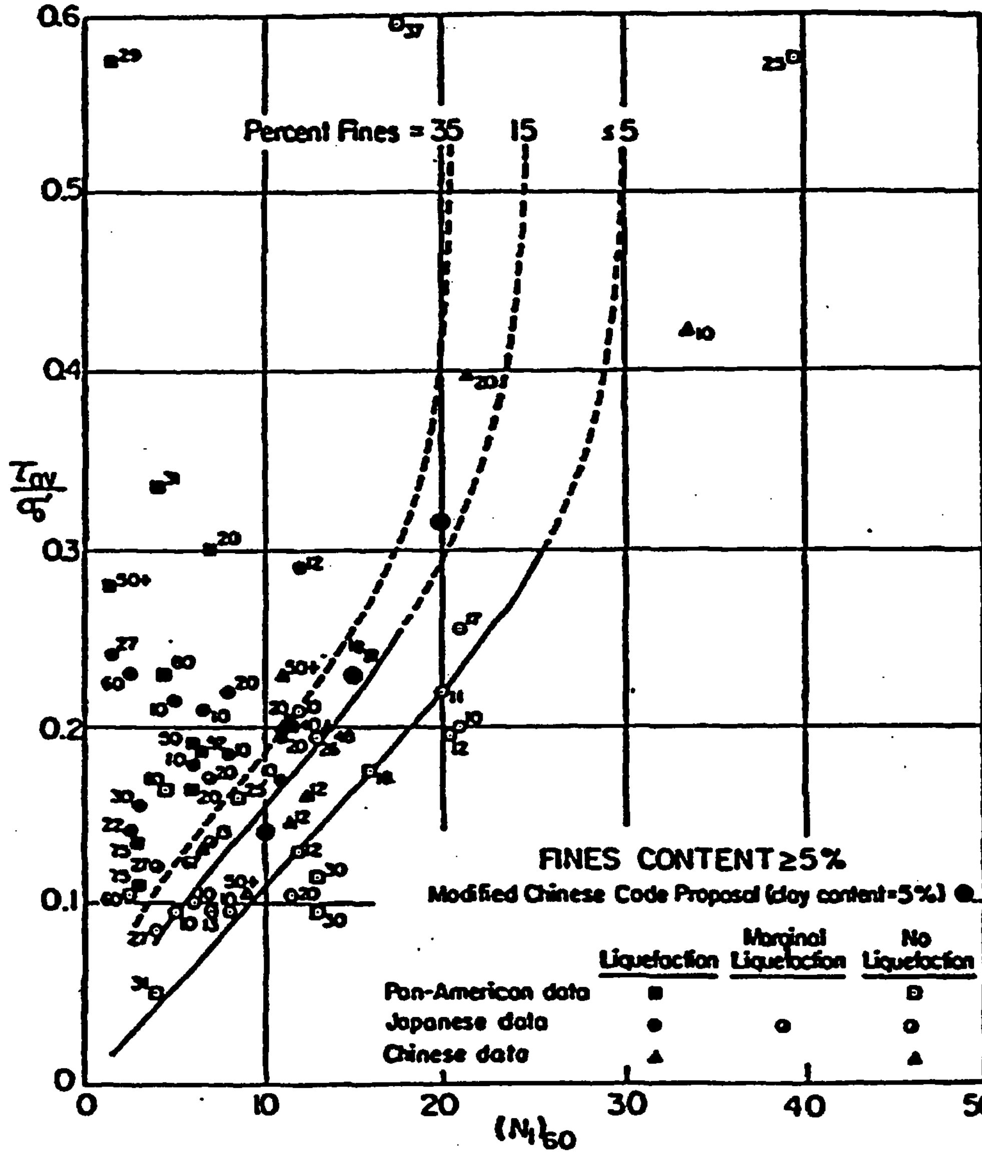
GEOMATRIX

SECTION B-B'
From FSAR

Figure
3

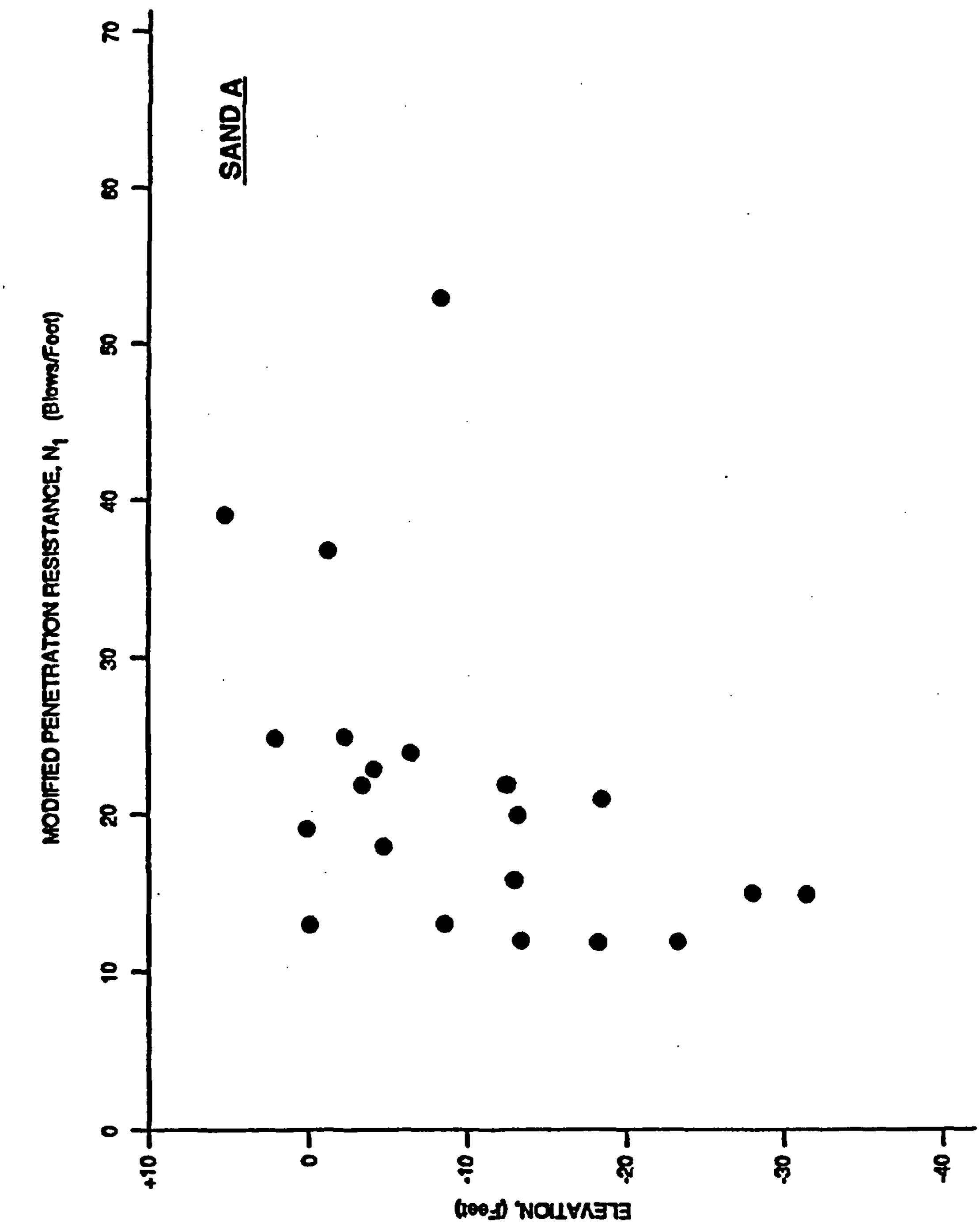
Project No.
1134A

F-19



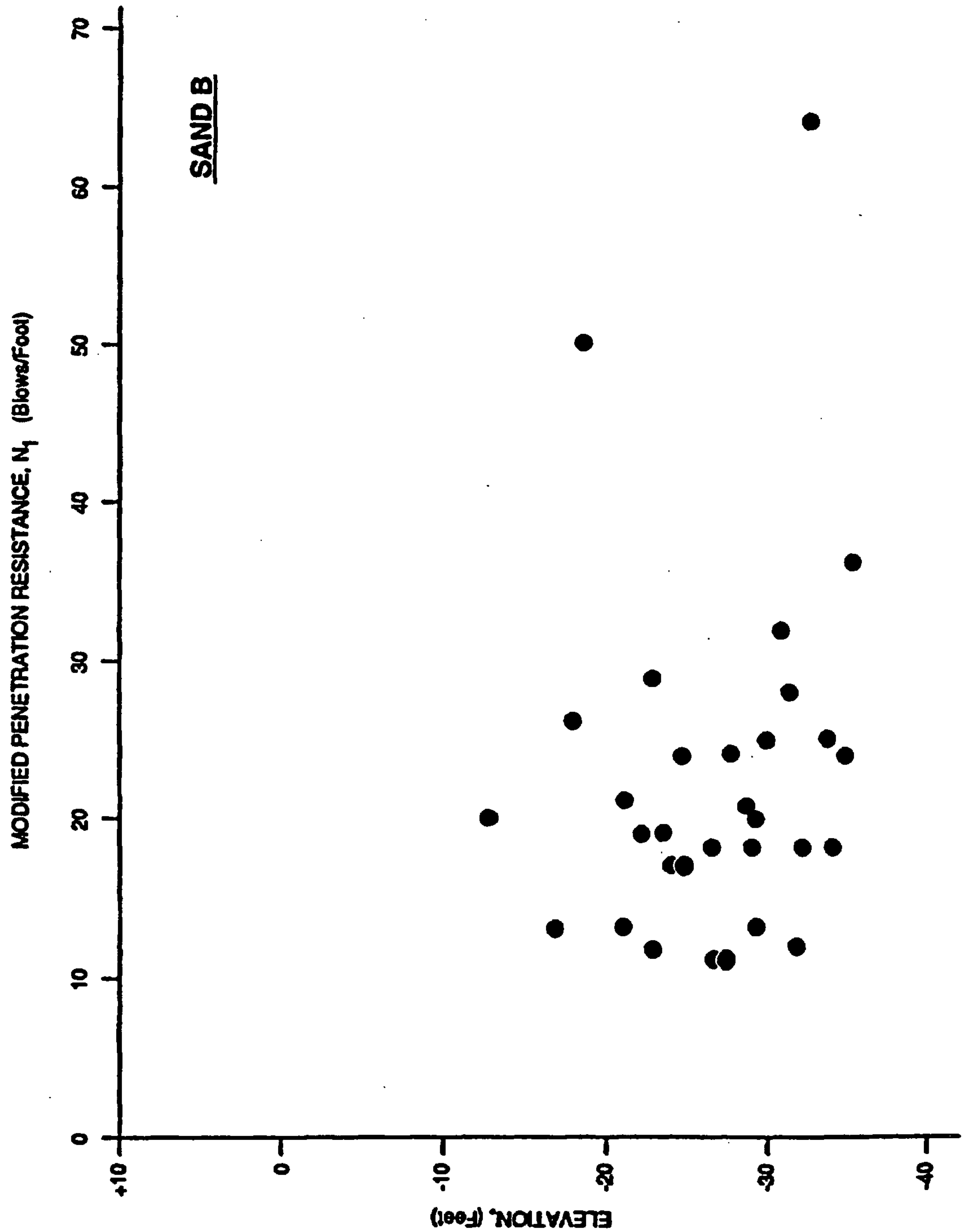
GEOMATRIX	RELATIONSHIP BETWEEN STRESS RATIOS CAUSING LIQUEFACTION AND N_1 VALUES FOR $M = 7\text{-}1/2$ EARTHQUAKES FROM SEED AND OTHERS, 1985	Figure 4
		Project No. 1134A

F-20



GEOMATRIX

GEOMATRIX	MODIFIED PENETRATION RESISTANCE VALUES VERSUS ELEVATION FOR SAND A	Figure 5
		Project No. 1134A



GEOMATRIX

MODIFIED PENETRATION RESISTANCE VALUES VERSUS ELEVATION FOR SAND B

Figure 6

Project No. 1134A

Review Comments of the SOARCA Project by Karen Vierow
April 16, 2010

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 US by this reviewer and other researchers. Multiple journal papers 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 accompany the "modernization" to a newer FORTRAN version, while 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 US codes for severe accident analysis. Comparing the different severe accident codes' predictions against experimental and plant data is an essential test of the codes' 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. 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 has enabled results which are more realistic than previous analyses. The severe accident calculations also include modeling improvements and insights which have been achieved since the earlier calculations were performed.

Some analysis aspects remain which require additional sensitivity studies and uncertainty quantification. Conservative safety factors have been applied in certain areas where uncertainty remains. As recommended in an April 9, 2010 memo to the SOARCA team, uncertainty quantification and sensitivity analysis are essential to the credibility of the SOARCA. Since the Peer Review Committee's charge does not extend to the uncertainty quantification and sensitivity analysis, further suggestions are not made herein. However, this reviewer believes that "closing the loop" on remaining issues via uncertainty quantification and sensitivity analysis will enable achievement of the SOARCA goals.

3 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.

4 Attainment of SOARCA Objectives

The SOARCA objectives are, quoting from the Executive Summary in the Summary Document:

The overall objective of the State-of-the-Art Reactor Consequence Analysis (SOARCA) project is to develop a body of knowledge regarding the realistic outcomes of severe reactor accidents.

Corresponding and supporting objectives are as follows:

- *incorporate the significant plant improvements and updates not reflected in earlier assessments including system improvements, training and emergency procedures, offsite emergency response, and recent security-related enhancements described in Title 10, Section 50.54(hh) of the Code of Federal Regulations (10 CFR 50.54(hh)) as well as plant updates in the form of power uprates and higher core burnup;*
- *incorporate state-of-the-art integrated modeling of severe accident behavior 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 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; and,*
- *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, 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 are 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. Consideration of power uprates and higher core burnup in the MELCOR analysis is unclear. The effect of higher burnup would be seen in the radionuclide inventories.

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 Appendices A and B, which present the comparisons of mitigated and unmitigated scenarios. Mitigation steps have large, positive effects on the event progression and consequence reduction.

The documents are thorough and well-prepared. Members of the public who are willing to invest time and have a familiarity with nuclear and other related technologies, will be able to understand the SOARCA approach and results as presented in the SOARCA document. The Executive Summary presents the four volumes of information in a concise format. For the general public who is less familiar with the technologies, documents written in layman's terms are 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.

5 Unaddressed Items, Future Work Items

5.1 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 Appendix B state that "This study has focused on providing a realistic evaluation of accident progression, source term and offsite consequences..."

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

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

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

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

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

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

5.2 MELCOR modeling of steam generator tube failure

Replacement of the SOARCA model for thermally-induced steam generator tube rupture at high pressure with a mechanistic model should be considered as a future work item. Thermally-induced steam generator tube rupture is deemed to occur in the SOARCA analyses when the cumulative creep damage index of one of the hot legs exceeds a value large enough to ensure that hot steam is passing through the steam generator tubes. Tube rupture is then imposed upon the calculation so that this is the first structural failure of the Reactor Coolant System pressure boundary.

A different approach which has been developed and documented by this reviewer is summarized below. Adoption of this or another mechanistic model for SOARCA analysis may be considered as a future work task, as opposed to a current SOARCA deficiency, for two reasons. Firstly, the SOARCA methodology appears necessary to assure that a thermally-induced tube rupture is the first structural failure in the event sequence. Secondly, the SOARCA team has performed further investigation into the short-term blackout with thermally-induced steam generator tube rupture to confirm that the hot leg would fail within close time proximity to the steam generator tube rupture(s).

The main benefit of a physics-based model would be that it is more defensible from a technical standpoint. The timing of Reactor Coolant System pressure boundary failures is close enough that other failures would occur before the primary system could significantly depressurize through the first failure, as demonstrated by both the reviewer's calculations and the SOARCA analyses. See, for example, SOARCA plots of the primary and secondary pressure responses for the unmitigated and mitigated 100% and 200% TI-SGTR STSBO events. The relative timing of the failures is important because the duration of the containment bypass at high pressure influences the source term release to the environment.

Liao and Vierow [2005] developed a method to estimate the steam generator hottest tube wall temperature and the tube critical crack size for the steam generator tubes to fail first. Best-estimate assumptions regarding the steam generator mixing parameters, steam generator hottest tube gas temperature adopted from CFD calculations and pressurizer void time adopted from a three-code comparison were applied to determine the critical crack sizes for the steam generator tubes to be the first failure in the Reactor Coolant System pressure boundary. However, nonuniformity in the gas temperature distribution among the steam generator tube bundle demands analysis of the hottest tube creep-rupture failure. Because of the limitation of one-dimensional codes, a prediction method was proposed to conservatively estimate the hottest tube wall temperature from the average tube temperature history, which is calculated by MELCOR. If the hot-leg nozzle thermal failure is considered, the tube critical crack size based on the hottest tube failure is about 40% of wall thickness smaller than that based on average tube failure.

Steam Generator Spatial Nodalization

The calculations by Liao and Vierow also included a more detailed spatial nodalization of the steam generator tubes. The SOARCA nodalization has a single control volume for each half of the U-tube. A best-estimate input deck should include a more detailed nodalization because this reveals the axial location of thermally induced failure (most likely to occur at the connection of the U-tubes to the tube sheet as assumed in the SOARCA analysis) and enables axial profiles of the fluid temperatures and small pressure differences which drive natural circulation.

6 Appropriateness of Presentation in the SOARCA Documents

6.1 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 implies a fair review of the process and makes possible an adequate and impartial evaluation of the SOARCA.

6.2 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 objectiveness 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 non-evacuees.

The MELCOR Best Modeling Practices volume 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 which 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 at least with the same level of fidelity as other leading severe accident codes in the US. Most of the models used in SOARCA have been validated against plant data and separate-effects test data.

Some analysis aspects remain which 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 of the Summary Document, have been achieved in large part. In particular, a large amount of information regarding severe accident analysis has been acquired. 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 could 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.

Suggestions were provided for a mechanistic model of steam generator tube rupture. While the current simplified model was necessary to enable a thermally-induced steam generator as the first structural failure, a mechanistic model would be easier to justify on a technical basis.

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.

Reference:

Y. Liao*, K. Vierow, "MELCOR Modeling of Creep Rupture in Steam Generator Tubes", Nuclear Technology, Vol. 152, No. 3, pp. 302-313, 2005.

Individual Assessment by Jacquelyn Yanch to be inserted here.

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.

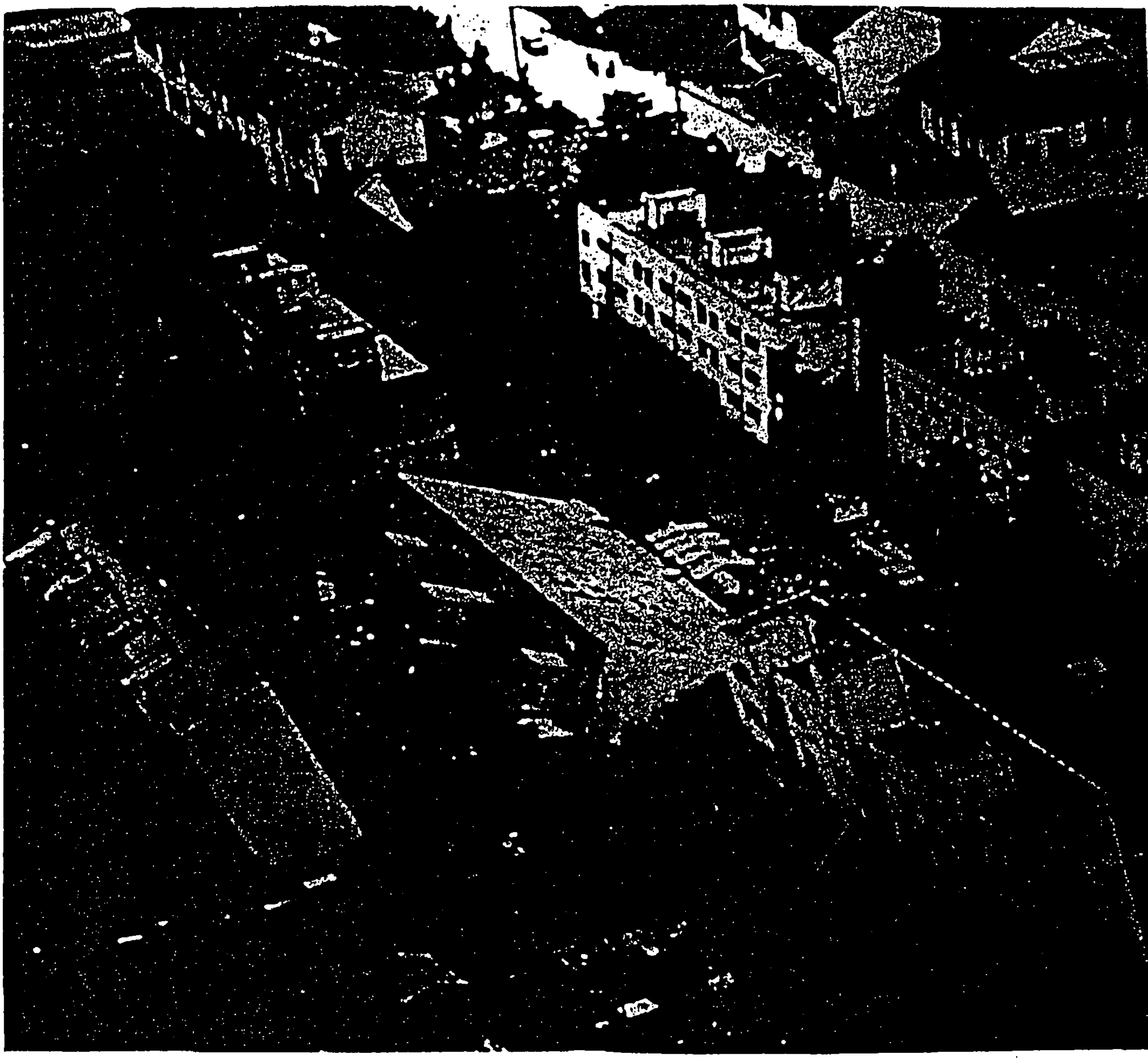


FIGURE 9 TILTING OF APARTMENT BUILDINGS AT KAWAGISHI-CHO DUE TO SOIL LIQUEFACTION RESULTING FROM THE NIIGATA EARTHQUAKE, JUNE 16, 1964.

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**	

No.	Page	Section	Type	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 Engineer<u>s'</u>	
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)?	

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**	
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.**	

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**	
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.	

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

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Date Comment Sent: 20 August 2009</td></tr>
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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 1×10^{-6} per reactor year has been chosen for those sequences groups that contribute to core damage and 1×10^{-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 1×10^{-8} per year. This is significantly lower than the SOARCA truncation limits.
   ii. LOCAs with vapor suppression failure which is also estimated at 1×10^{-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.*

<u>*Jeff Gabor supports inclusion of Jacquelyn Yanch's comments in the SOARCA documents.*</u>

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

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.

<u>*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.*</u>

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.

<u>*Kevin O'Kula provided the following comments on Federal Guidance Report 13.*</u>

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.*

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

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.

<u>John Stevenson recommends an evaluation of plant foundation conditions</u>.

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
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