SOARCA
Peer Review
DRAFT Comments

May 4, 2010

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2/22

Clement - re uncertainty analysis - parameters not fully independent, whereas they should be for MC.
answer - some parameters must be correlated, or else physically-unrealistic regions may be encountered. Several of these are discussed in the documentation of the US/CEC uncertainty study, with the reasons for correlating. For example, dispersion parameters must be correlated, or the situation may be encountered where dispersion is larger for stable conditions than for unstable conditions
re tails of infinite distributions, functions other than normal distributions should be considered.
The off-site parameters for the US/CEC study have considered these issues.

provides representative results for selected SA sequences -

Gabor no L3-specific issues discussed, but he notes the limited scope of SOARCA re risk
has L2 PRA in wish list, but references "risk" in the sentence. Does he mean L3?
Henry no L3 - specific issues discussed
Kowieski re Don't show results to 100 miles
Note that the metric has been changed compared with the previous studies. Therefore, while the preceding metric, total LCF, increases in magnitude with increasing distance, the present metric, individual risk of LCF, becomes smaller in magnitude with increasing distance. The SOARCA, with the restriction to 50 miles, is showing larger values. In the peer reviewer Evaluation of the response section, he states that much of the particulate material would have been deposited within about 50 miles. While we don't have an edit that specifically addresses this, it probably IS NOT TRUE. Of course "much" is not defined

2

re non-evacuating public for SURRY - (concept not needed for P.B.)

100% of the emergency phase risk is from the non-evacuees is true for the EPZ. We need to check to make sure that it is true for beyond the EPZ. Further, the concept that it's their own fault is not a viable argument - I believe in law, there is the concept of contributory negligence, but I don't think it goes this far -

Further work in L3 PRA may be beneficial for confirm purposes

Leonar re previous studies excessively conservative New associated risk greatly reduced.

ST is very much lower and delayed in time - Robust conclusion - therefore EF significantly reduced - BUT there were non-conservatisms in previous studies in the off-site consequence calcs, that have been corrected. Because of many changes, there is NO CLEAR comparison with LCF for the previous studies, but it is suspected that there is not much change. Certainly NOT "greatly reduced"

Mrowca no L3 but questions the L1 conclusions and also the all scenarios could reasonably be mitigated!

Full L3 NOT needed.

O'Kula re need to document MELMACCS code and process

update documentation -

re compare to NUREG-1150 -

XSOR code process makes the comparison extremely difficult - 1150 made only limited use of individual risk results (to compare with Safety Goals) - other metrics difficult to compare EF significantly less, mainly probably due to delay in ST. LCF may not be much less -

re uncertainty studies - No food pathway - table of suggested parameters -

will consider when the time comes.

re surface roughness - maybe "overly conservative" for Peach Bottom.
ask Nate -
re aerosol particle size + deposition velocity - add to documentation - The US/CEC study evaluated NUREG notes that quantile is more important than surface roughness -
re clarify The radionuclide assignment to aerosol settling velocity -
clarify document re source term has different fractions of chemgroups in different bins for different plume segments come from MELCOR through MELMACCS -
re Need tables of PB + Surry input values add more to the documentation but also include CD
re Boundary weather -
clarify that boundary weather was not used within the reported 50 mile results -
re reporting land contamination, 50-mile dose - decision not to do ~~land~~ contamination possibly will do for uncertainty study - also, with changes in FGR values and organs, comparison is difficult -
re 100 mile values - no regulatory basis 0-50 mile value larger than 0-100, so we are reporting the larger values.
re documentation changes -
① tables of LCF/event + LCF/reactor-year good idea
② table acute + longterm good idea
③ Select 2 of the 4 models - not good idea - who clue as to why he suggested this -
④ Shorthand notation for dose - response results good idea -

Vierow re non-evac cohort clearly ID'ed as voluntary non-evacuees -
already in documentation, but could be made stronger -
Yonch re we know little about low dose health effects - and low dose-rate -
Tried to reflect the various opinions in the national and international community by giving results assuming 4 dose-response models -
re what is the health impact of the habitability criterion dose? Don't know
a ques - DOE has a program looking at this very issue -
re public bears the burden of up heaval - ~~would~~ major dose-avoidance strategies (such as, long term relocation until the habitability criterion is met) unlikely to be in the public's best interest
DHS has optimization strategy to be applied by the local/state after an incident to balance response
re Estimated dose-response from the US/CEC study - NUREG-6555 (?)
Need to check - don't recognize the reference - we used values from Keith Eckerman
ah elsewhere is NUREG/CR-6555 - we did not use this. used BEIR V via K.E. and DDREF = 2.0
used 4 truncation doses.
re recommendations
none under the purview of NRC
re March 2010 memo - use BEIR VII?
no comparable work to FGR yet for the tissues in BEIR VII. they may or may not be the same as in FGR-13. EPA (Keith Eckerman, ORNL) is working on this now - will reevaluate for future work when he is done -

(5)

Table comments re why not do METCOD = 5 NSAMPLES = 24

while LNT runs are fast and 8760 mets samples not much longer than what we are doing already, the threshold value runs take a lot longer. There probably is not a lot to be gained from further met sampling compared with other sources of uncertainty, met's adequately treated now -

re deposition velocity -

added sentence to indicate that the effect is to increase ACF by ~~about~~ 3 - agree that it is much more important than most people give credit for -

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

Manuscript Completed: *(expected May 15, 2010)*

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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.

we surely got this message out

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.

But we also corrected the excessive non-conservatisms

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.

a curb

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.

This | yes, but is what [illegible] to downplay and state [illegible] of [illegible] the [illegible] that [illegible] represented.

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

RC

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.

all too true!

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 term analyses. For the same reason, more input will be found for the Surry PWR than for the Peach Bottom BWR.

SOARCA Objectives and Approach

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

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

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

For fully answering the question: "is SOARCA a best-estimate study" one needs to consider the accident scenario selection procedure. 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.

all too true – The "stop" of all releases by flooding over the debris would be a bad assumption if this is true, and I expect it is.

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. good statement

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. good statement!

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.

Perhaps it would be ok if the limitations were more clearly stated, the sensitivity attempts to answer the needed that SOARCA also captures the majority of the risk

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.

Level 3!!
?

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 describe~~s~~ 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. Last page says 99.8%
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. Not a viable stance within the EPZ

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.

There are some claims that most of the risk is captured -

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.

LCF true
EF false

no, not examples — only a guidestone argument (applicable only to LCF) that there could not be scenarios with 10X the release to offset 1/10 in CDF

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.

NA applicable to EF

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

true

not true for LCF

true for EF+ release fraction NOT true for LCF due, in part, to excessive non-conservatism in that model

near zero

risk factors a/ dry deposition velocity

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.

State-of-the-Art Reactor Consequence Analysis (SOARCA) Project - Independent Technical Review

Bruce Mrowca

~~DRAFT~~

Review Objective

as always the L1 PRA folks. Change things so all there is for PRA risk assessment!

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 sequence~~s~~ only by-pass — with a core damage frequency (CDF) of greater than 10^{-6} per reactor-year. In addition, seque~~nces~~ 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. — this document was not a quantification of the risk

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. No, we don't want to do this
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.
   CDF? until the whole analysis is done we don't have the risk profile
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. Good point
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.

URS

FILE NO.	URS SMS-SAE-M-10-0007	DATE:	May 17, 2010
		LOCATION:	URS SMS LLC/Aiken

TO: **Professor Karen Vierow**
Department of Nuclear Engineering
Texas A&M University

FROM: **Kevin O'Kula**

CC:

SUBJECT: **State-of-the-Art Reactor Consequence Analysis (SOARCA) Project Assessment**

Introduction and Summary

An assessment was conducted as part of the State of the Art Reactor Consequence Assessment (SOARCA) Peer Review Team's review of the methods, inputs, analyses, results and findings for Peach Bottom and Surry Nuclear Power Plants for selected severe accident sequences. The SOARCA review was based on Peer Review meetings in Rockville, Maryland, and SOARCA documents. Peer Review meetings were held in July and September of 2009, and also in March of 2010. Several drafts of the four-volume SOARCA (NUREG) report (Ref. 1) have been made available to the Peer Review Team along with numerous supporting documents of a precedent or contemporary nature.

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

Section 1 - Adequacy of the SOARCA Concept
Section 2 – SOARCA Approach
Section 3 – Reasonableness of the SOARCA Technical Results
Section 4 – NUREG Approach and Uncertainty Analysis
Section 5 – Attainment of the SOARCA Objectives
Section 6 – Unaddressed Items and Opportunities for Improvement, and
Section 7 – Appropriateness of Presentation in the SOARCA Documents.

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

1. Adequacy of the SOARCA Concept

In general, the SOARCA study applied valid approaches for evaluating severe accident phenomena and the subsequent offsite consequences. With respect to offsite consequence analysis, many parts of the analyses could be regarded as "state-of-the-art, including:

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

However, other aspects of the offsite consequence analyses maintain older models or input data. These are judged to be adequate for achieving the overall goals on the SOARCA project in most cases but merit revision at the earliest opportunity. These include:

- Straight-line Gaussian model
- Economic consequence model with older (e.g. NUREG-1150, Ref. 2) models, assumptions and input data on decontamination and recovery of economic assets.
- Limited, if any, use of the uncertainty capability in WinMACCS – perhaps this will be utilized in the subsequent uncertain analysis but this new feature of the software was apparently not exercised in the base case work reported to date.

← asks for code change - what other code would he suggest?

2. SOARCA Approach

In an overall sense, the SOARCA project, practices and methodologies are described sufficiently, with appropriate level of detail for the accident scenario selection process. Criteria for event selection and screening are defined sufficiently and appropriate for the intent of the SOARCA analysis.

For the Peach Bottom plant, this process resulted in the following sequences

- Long-term station blackout
- Short-term station blackout with RCIC blackstart
- Unmitigated short-term station blackout, and
- Unmitigated short-term station blackout, accounting for seismic activity.

For the Surry plant, the SOARCA process resulted in the following sequences

- Unmitigated short-term station blackout
- Unmitigated short-term station blackout with thermally induced steam generator tube rupture
- Mitigated short-term station blackout with thermally induced steam generator tube rupture
- Unmitigated long-term station blackout
- Unmitigated interfacing systems loss of coolant accident, and

- Sensitivity studies with the unmitigated interfacing systems loss of coolant accident.

The discussion presented during Peer Review Team meetings and in the documentation was sufficient to justify using a set of robust scenarios for each plant rather than aiming for completeness and adding other scenarios that contribute marginally to total plant risk.

value judgement? what is the definition?

how does the know that omitted sequences would only add marginally to risk?

Adequacy of the current Level I PRA for the two plants under consideration was not assessed. We accepted point mean values for accident sequence frequencies as valid.

It is the opinion of this reviewer that a full Level III PRA is not needed to gain key insights at this phase in the process. Information is not available to have taken either plant to this extent, and the level of resource commitment to perform a complete Level III analysis would not have been commensurate to the additional information gained. I would rather see other type containments analyzed through the same SOARCA processes for both PWRs and BWRs, and representative of the current U.S. fleet.

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

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

There was limited work done elsewhere in several areas, and could be but these are discussed elsewhere in the present assessment.

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

we can add more documentation

The SOARCA processes were sufficiently best-estimate with respect to the offsite consequence analysis performed. It will be important to perform a reasonable uncertainty and sensitivity analysis to understand parameter and model sensitivities, and where the best-estimate values lie relative to other quantiles.

3. Reasonableness of the SOARCA Technical Results

In general, the overall technical results are well substantiated and explained in sufficient detail so as to support key findings and study insights. While good use is made of the NUREG/CR-2239 (Ref. 3) SST1 source term with respect to the composition, timing, and magnitude of the release relative to SOARCA source terms, the opportunity should be seized to connect with Peach Bottom and Surry results from NUREG-1150 where practicable. The SOARCA study is an opportunity to build on the discussion from the landmark severe accident risk study for Surry and

Not practicable given the XSOR process

only of BP

Peach Bottom to show how improvements in methods, training, modeling, plant improvements, have substantially reduced severe accident risks. This comparison would be highly informative for those accident sequences, e.g., station blackout or LOCA that were analyzed throughout the 1970s-1980s and have been revisited during the SOARCA study.

4. NUREG Approach and Uncertainty Analysis

Because the SOARCA project is a best-estimate realistic approach to severe accident analysis, it is important to perform at least a limited uncertainty analysis. Several aspects of planned uncertainty quantification and sensitivity analysis, along with the advantages and disadvantages of two techniques to account for weather variability, were discussed with the Peer Review Panel in early March. In lieu of a Parameter Importance and Ranking Table process, it is our understanding that the judgment of SNL-NRC SOARCA team and the Peer Review Panel will be used to identify important parameters for review.

I hope not

PIRT-like process was done for non-site specific parameters prior to the US/CEC study-

The parameters recommended for follow-up consideration in the uncertainty study are listed in Table 1. The phase of analysis indicates whether the parameter is from the accident sequence or offsite consequence analysis (first column). The subject indicates the topical area that the parameter affects (second column), and the parameter is given in the third column.

we have not established such goals yet!

It is this reviewer's judgment that a side study on uncertainties associated with the food or water ingestion pathway is not prudent, and falls outside the goals of the present uncertainty analysis. Given that the SOARCA study aims to be a realistic assessment of severe accidents and their aftermath, the assumption that foodstuffs into the affected region would be from uncontaminated areas, eliminating this pathway is valid and there is not any value to be gained from exercising food pathway module required for this purpose. but even if we set the dose for food to zero, we need the pathway for cost.

he would set the DOB to zero - I would give the food pathway a very small, but not 0, value

Table 1. Recommended models/parameters for Uncertainty Analysis in the SOARCA Study

all these surely on the uncertainty list -

Phase of analysis	Area	Parameter/Model
Accident sequence	Core degradation	Fission product release from the fuel
Accident sequence	Radionuclide chemistry	1. Iodine and Cesium chemical forms 2. Barium chemistry 3. $CsMoO_4$ formation and chemistry
Accident sequence	Operator actions	Mitigative action timing
Offsite consequence analysis	Atmospheric transport and dispersion	1. Surface roughness length, z_o 2. Deposition velocity
Offsite consequence analysis	Long-term dose mitigation	1. Population dose return criteria 2. Decontamination efficacy and cost 3. Land and economic asset values

others should probably be added!

5. Attainment of SOARCA Objectives

It is the judgment of this reviewer that the SOARCA project largely met its over-arching objective as stated in the SOARCA Summary Document, Revision 1, of 14 February 2010, i.e., "to develop a body of knowledge regarding the realistic outcomes of severe reactor accidents". Included with the primary objective are corresponding and supporting objectives. While many areas are still being addressed and need additional work, these include: (i) incorporating the significant plant improvements and updates not reflected in earlier assessments including system improvements, training and emergency procedures, offsite emergency response, and recent security-related enhancements described in Title 10, Section 50.54(hh) of (10 CFR 50.54(hh)) as well as plant updates in the form of power uprates and higher core burnup; (ii) crediting state-of-the-art integrated modeling of severe accident behavior which includes the insights of some 25 years of research into severe accident phenomenology and radiation health effects; (iii) evaluating the potential benefits of recent security-related mitigation improvements in preventing core damage and reducing an offsite release should one occur; (iv) enabling the NRC to communicate severe-accident-related aspects of nuclear safety to stakeholders; (v) updating quantification of offsite consequences found in earlier NRC-sponsored work such as NUREG/CR-2239.

already included in (i) above!

Remaining sections discuss specifics on *Unaddressed Items and Opportunities for Improvement*, (Section 6) and *Appropriateness of Presentation in the SOARCA Documents*, (Section 7).

6. Unaddressed Items and Opportunities for Improvement

There are several items that merit attention before the NUREG is published. Most of these are viewed to be relatively minor and would be dispositioned with additional discussion.

1. **MELCOR-to-MACCS2 transition** - The documentation in the four-volume NUREG report, and especially in Volume 1, Summary report, is sparse with respect to the MELCOR to MACCS2 transition. It is difficult to judge how best-estimate aspects of the source term description are based the description provided for deposition velocity and surface roughness length. ?

   In the Summary report, the discussion (pages 60-61) is ambiguous regarding the approach to assigning deposition velocity to aerosol particle sizes. Specifically, it is unclear if binning associated with particle size and deposition velocity uses expert elicitation and MELMACCS methodology or if one approach was primary and the other supplementary. The pedigree of the MELMACCS technical report (Ref. 47) appears to be at an internal laboratory report. It is recommended that the report be formally released with adequate technical review.

2. **Surface roughness length (z_0)** – The documentation in the Volume 1 Summary is very brief and omits much technical justification for selection of key parameters. One of the areas that remains too limited is the basis for the surface roughness length (z_0). Ten centimeters surface roughness length would seem to be overly conservative for Peach Bottom given the fifty-mile environment surrounding the plant. The same value may be appropriate for Surry but a more complete discussion is recommended. Is the 10-cm value used for both plants an indication that the environments around the plants were considered characteristic of tall grass (refer to Table 2.3 from NUREG-4691 Vol. 2 and shown in Figure 1 below)?

is less conservative a larger z_0? or a smaller z_0?

Table 2.3

Approximate Surface Roughness Lengths (z_o) for Different Surfaces

Type of Surface	z_o(cm)
Lawns	1
Tall grass, crops	10 - 15
Countryside	30
Suburbs	100
Forests	20 - 200
Cities	100 - 300

Figure 1. Representative surface roughness lengths for various types of surface (Volume 2 of MACCS Model Description, page 2-16 of NUREG/CR-4691; Ref. 4).

3. **Deposition velocity selection** - The deposition velocities associated with various aerosol sizes is given in Table 13 of the Summary (page 61). However, it is not clear what radionuclides groups are associated with one or more of the bins shown in Table 13 and how the median diameter bins would be distributed for a given radionuclide group. Surely a state-of-the-art input distribution of deposition velocities would offer a different distribution based on physicochemical characteristics of one group vs. those of another. For example, the halogen isotopes (primarily radioiodine) will deposit differently from cesium species (e.g., CsI and $CsMoO_4$). It is suggested that while realistic input is reflected in the SOARCA study and used in the current analyses, justification should be provide to support its use in place of the sample input published in the 1998 MACCS2 User's Guide (Ref. 5) for the allocation fractions for nine fission product groups (Figure 2).

Source term documentation

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

Figure 2. Allocation fraction input illustration from page 5-27 of MACCS2 User's Guide

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

4. **Discussion of the non-site specific and site-specific parameters used**: A key outcome of the SOARCA project, discussed several times with the Peer Review Panel during our meetings, has been the smaller and delayed source term compared to the Sandia Siting Study. These important results are illustrated in Table 21 (Appendix A – Peach Bottom Integrated Analysis) and Table 32 (Appendix B - Surry Integrated Analysis). Especially in the case of the Surry conditional risks with distance (Figures 172, 174, 176, 178, and 180), the results are dominated by the long-term (CHRONC) phase of the analysis. This aspect of the model is driven by user input of dose criteria (habitability dose limits), site-specific data, and some non-site specific information. It would improve the understanding of the SOARCA study to provide thee tables of information to augment the discussion of sections 5.5 and 5.6. The tables would mostly address the type of inputs that control the EARLY and CHRONC modules, specifically in deciding on condemnation and inhabitation return and be as follows (with suggested location by volume):
   - Table 1. Peach Bottom Offsite Consequence Analysis Site-Specific Parameters (Appendix A)
   - Table 2. Surry Offsite Consequence Analysis Site-Specific Parameters (Appendix B)
   - Table 3. Site-Independent Parameters (Summary - Section 5.6).

   This is why I want a disk!

   While some of these data (e.g. shielding factors) can be found elsewhere in the four-volume set of SOARCA documents, it would assist the reader to see the key inputs collected in one or several tables. This information would be useful in determining what sensitivity studies are important, what the focus should be on plant features, severe accident mitigation procedures, etc., and to what extent public policy has a role.
5. **Boundary weather** – During one of the review meetings, it was indicated whether a boundary weather condition is imposed, with forced deposition conditions, and if so, the type of weather specified and at what region in the grid. I don't think this is established in the February final draft for review document. This aspect of the offsite consequence analysis should be described for comparison with past work (e.g., NUREG-1150). documentation
6. **Centralized discussion of MACCS2 improvements**- There are many improvements noted in the MACCS2 model with 64 directional sectors and more realistic evacuation modeling among others. It would be informative to have a short section in Appendix A and Appendix B to summarize the prominent features and expanded capabilities by module, i.e., ATMOS, EARLY, and CHRONC. This version of the code has expanded capabilities for performing uncertainty analysis but little is covered in the documentation or was commented upon in the three presentations.

   Not modeling change, only user friendly front end – But new slow/speed in grid + slow for precip by hour
7. **Reporting of additional consequence measures** – In addition to the conditional and absolute early/latent health effect risks reported in the SOARCA study, other metrics would be advised. The uncertainty quantification and sensitivity analysis presentation in March 2010 indicated that land contamination was being considered, and would be very useful. It is advised to clarify whether forced deposition is used, and if so, the inner and outer radii the feature is employed. To compare to earlier studies, the metric of population dose over the fifty-mile region would be a useful consequence measure to complement land contamination. This is appropriate because deposition with distance would trend inversely with inhalation doses.

I prefer cost, but land contamination in addition is OK –

problem choosing a single nuclide + concentration

8. **Extent of Polar Grid** – Currently, the results presented in Appendices A and B for Peach Bottom and Surry, respectively, extend to fifty miles. The Summary document suggests that results will be reported out to a 100-mile extent. There is a regulatory basis for the 50-mile grid, including Ingestion Planning Zone, Environmental Impact Statement, and SAMA support analysis among others. A similar justification does not exist for a 100-mile basis, and it is recommended that the Summary be corrected to a 50-mile description.

7. Appropriateness of Presentation in the SOARCA Documents

In general, the SOARCA study appears objective and not noticeably influenced by licensees or other constituents. To improve the likelihood that the public will interpret the SOARCA study as intended several recommendations are made:

1. Aim for uniformity and consistency in the labeling of conditional and absolute risk figures and tables.- we suggest conditional risk per event (LCF/event) and for absolute risk (LCF/reactor-year). Currently, both types of risk are labeled the same in figures and tables. This situation may lead to incorrect interpretations by the reader. good point
2. Label LCF bar charts with Acute phase and Long-term phase rather than EARLY and CHRONC. The analysis is using MACCS2 as a tool to evaluate the relative importance of the short-term and long-term phases, and it should be made transparent that this is the case. Use of MACCS2 terminology in the results gives the appearance that the results are more characteristic of the manner in which the code was run, and not reflective of the post-release phases. why?
3. Select two of the four health effect (dose truncation) models rather than present results from all four models. The LNT model appears to be bounding in all cases. It is recommended that this model be retained along with the one that tends to predict the least health effect risks of the three alternative dose truncation models, i.e.,
   - Health Physics Society recommendation (5 rem/year and 10 rem lifetime)
   - ICRP Report 104 (10 mrem/year)
   - U.S. Average Background (620 mrem/year).

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

References

1. State-of-the-Art Reactor Consequence Analysis (SOARCA) Project, NUREG-1935, Predecisional Draft, February 2010. Summary, MELCOR Best Practices, Appendix A, and Appendix B.
2. NUREG-1150, V., Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants. 1990, Nuclear Regulatory Commission: Washington, DC.
3. NUREG/CR-2239, Technical Guidance for Siting Criteria Development. 1982, Sandia National Laboratories: Albuquerque, NM.
4. H-N Jow, J.L. Sprung, J.A. Rollstin, L.T. Ritchie, and D.I. Chanin. *MELCOR Accident Consequence Code System (MACCS), Volume 2. Model Description;* Sandia National Laboratories, Albuquerque, NM. NUREG/CR-4691 (SAND86-1562), 1990.
5. D.I. Chanin, M. I. Young, and J. Randall. *Code Manual for MACCS2: Volume 1, User's Guide*; NUREG/CR-6613 (SAND97-0594), Sandia National Laboratories, published by the U.S. Nuclear Regulatory Commission, Washington, DC, 1998.

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.

might truly but not - germane here

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

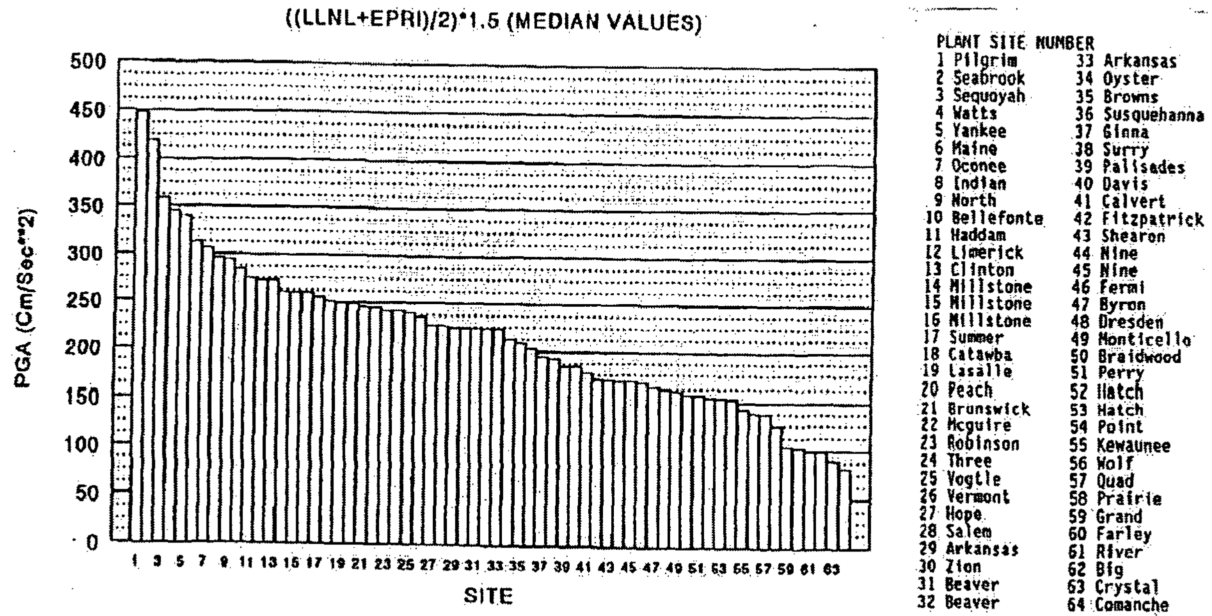

Figure 2 Median Peak Ground Acceleration at the 10^{-4}/yr Probability of Exceedence Level

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.

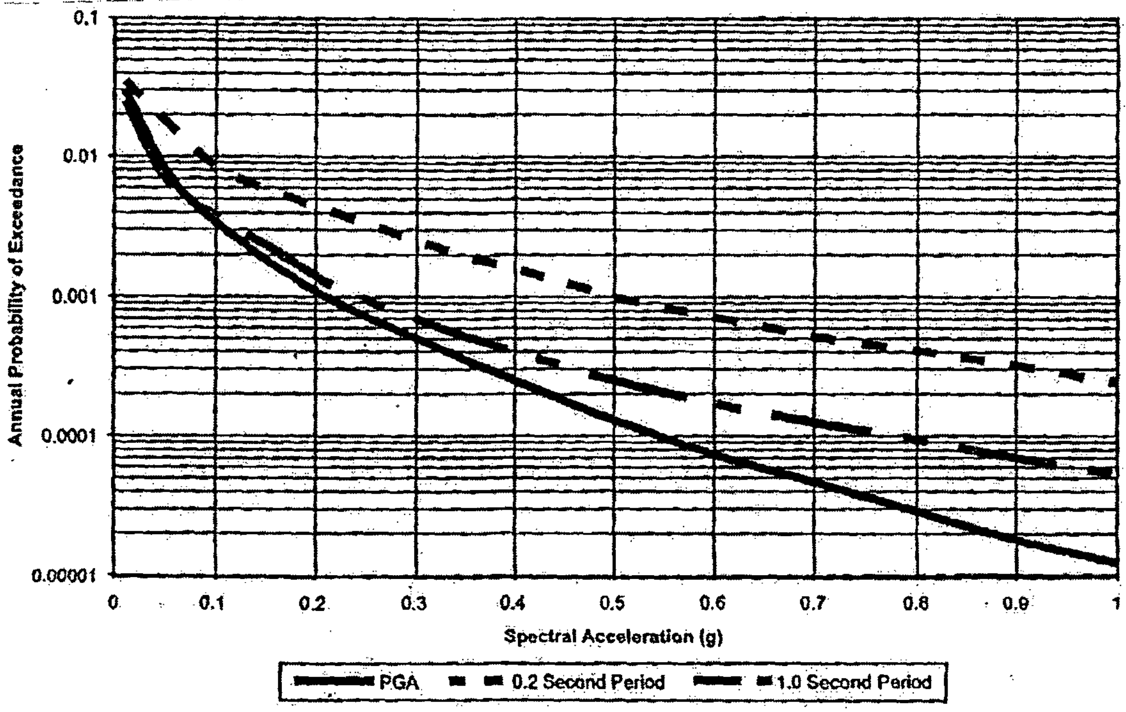

Figure 3 Typical Hazard Curve Taken From Figure 2-1 of ASCE Standard 43-05

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:

Sand A: 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.

Sand B: 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.

Sand C: 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).

Select fill: 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 Liquefaction Resistance of Sand C

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 Liquefaction Resistance of Select Fill

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:

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

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_B, at the foundation-soil interface.

a_B = 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_B, transmitted by the structures and the acceleration at the base of the structures, a_B, 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.

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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.

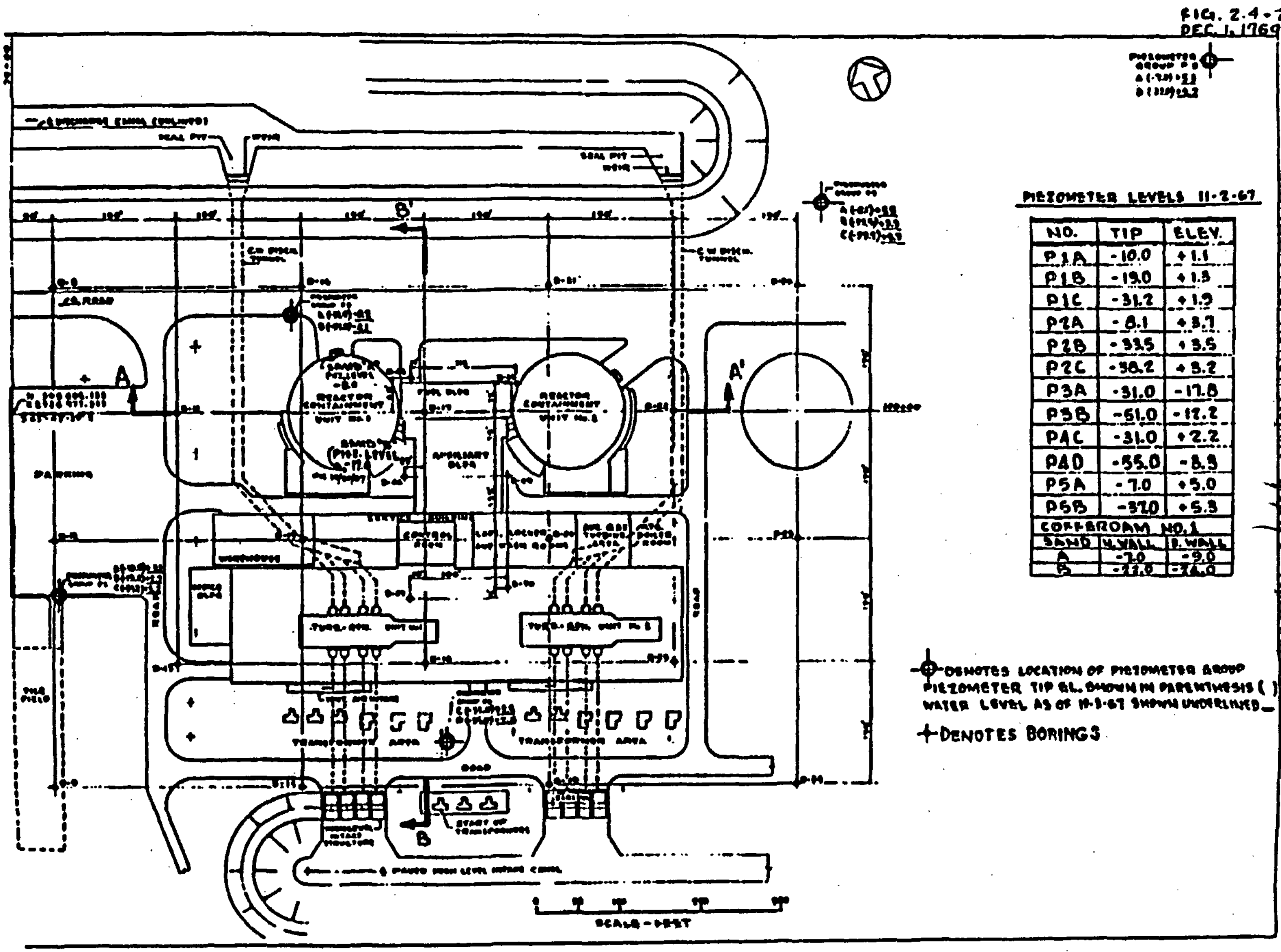
Tokimatsu, K. and Seed, H.B. (1978) "Evaluation of Settlements in Sands due to Earthquake Shaking ", Journal of Geotechnical Engineering, ASCE 113 (8): 861-878.

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.



PIEZOMETER LEVELS 11-2-67

NO.	TIP	ELEV.
P1A	-10.0	+1.1
P1B	-19.0	+1.5
P1C	-31.2	+1.9
P2A	-8.1	+3.7
P2B	-33.5	+3.5
P2C	-30.2	+3.2
P3A	-31.0	-17.8
P3B	-51.0	-12.2
P4C	-31.0	+2.2
P4D	-55.0	-8.3
P5A	-7.0	+5.0
P5B	-32.0	+5.3
COFFERDAM NO. 1		
SAND	W. WALL	E. WALL
A	-7.0	-9.0
B	-27.0	-26.0

⊕ DENOTES LOCATION OF PIEZOMETER GROUP
PIEZOMETER TIP EL. SHOWN IN PARENTHESIS ( )
WATER LEVEL AS OF 11-2-67 SHOWN UNDERLINED

+ DENOTES BORINGS

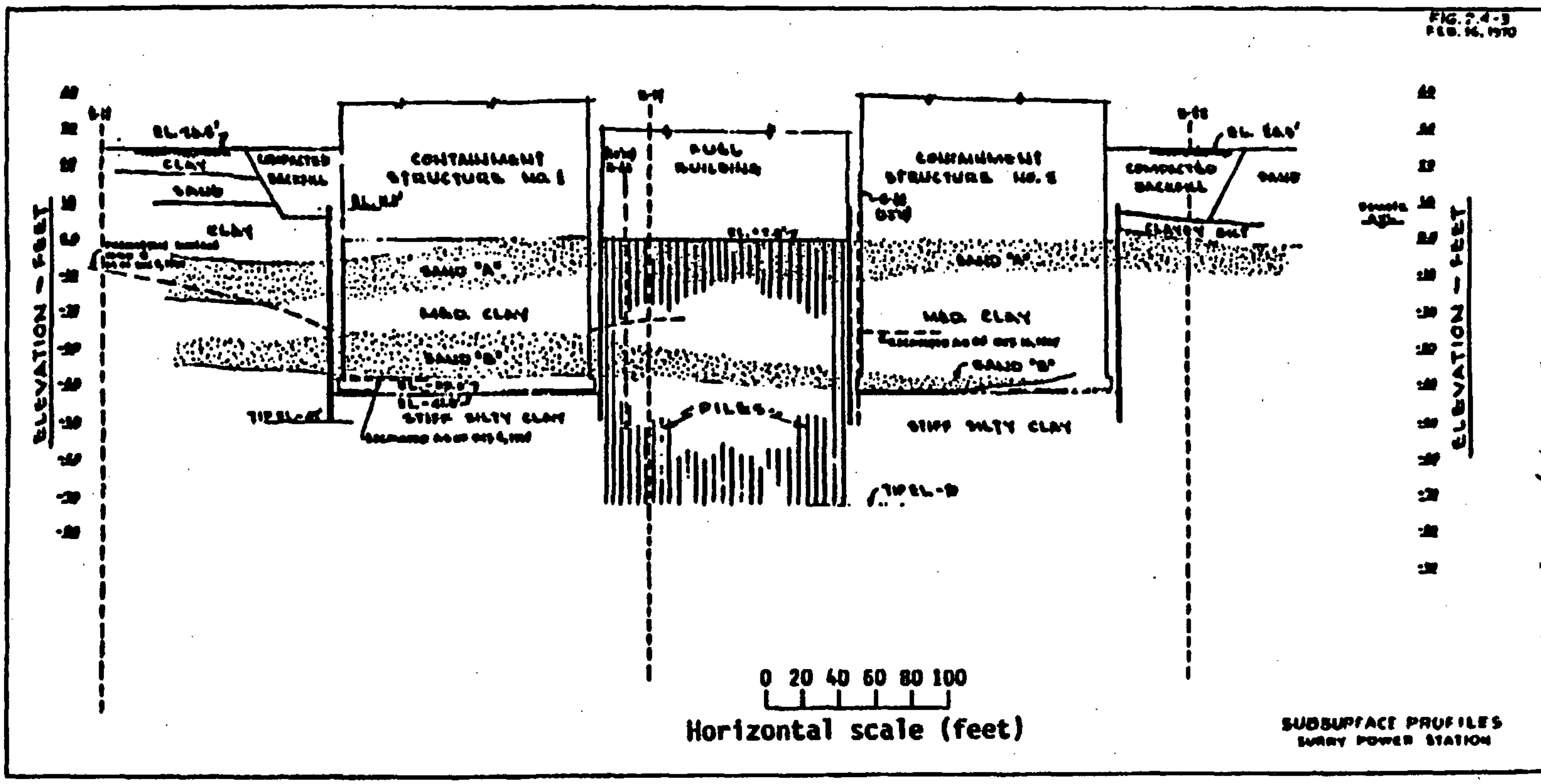
GEOMATRIX

PLAN OF POWER PLANT
From FSAR

Figure 1

Project No. 1134A

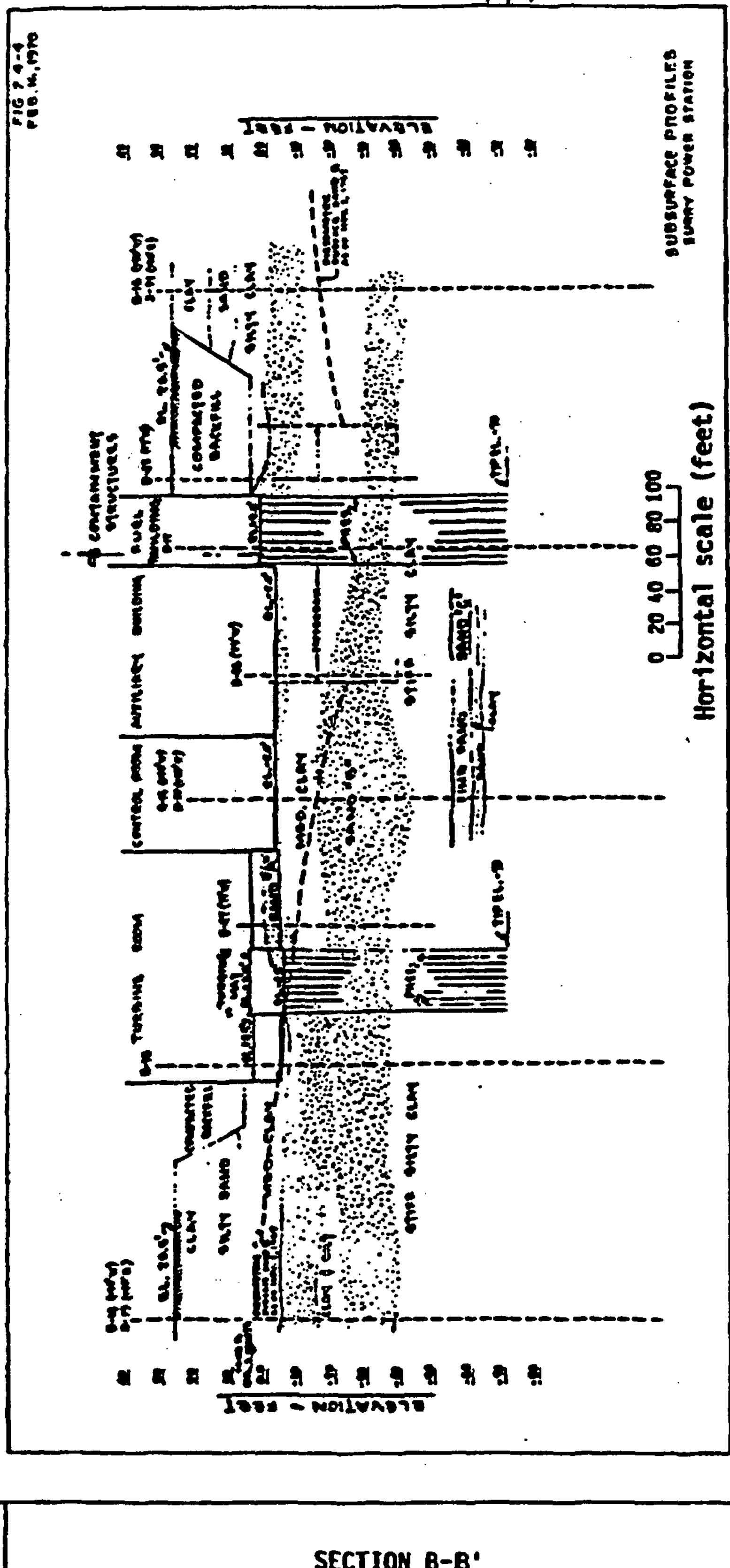
GEOMATRIX



SECTION A-A'
From FSAR

Figure 2

Project No. 1134A

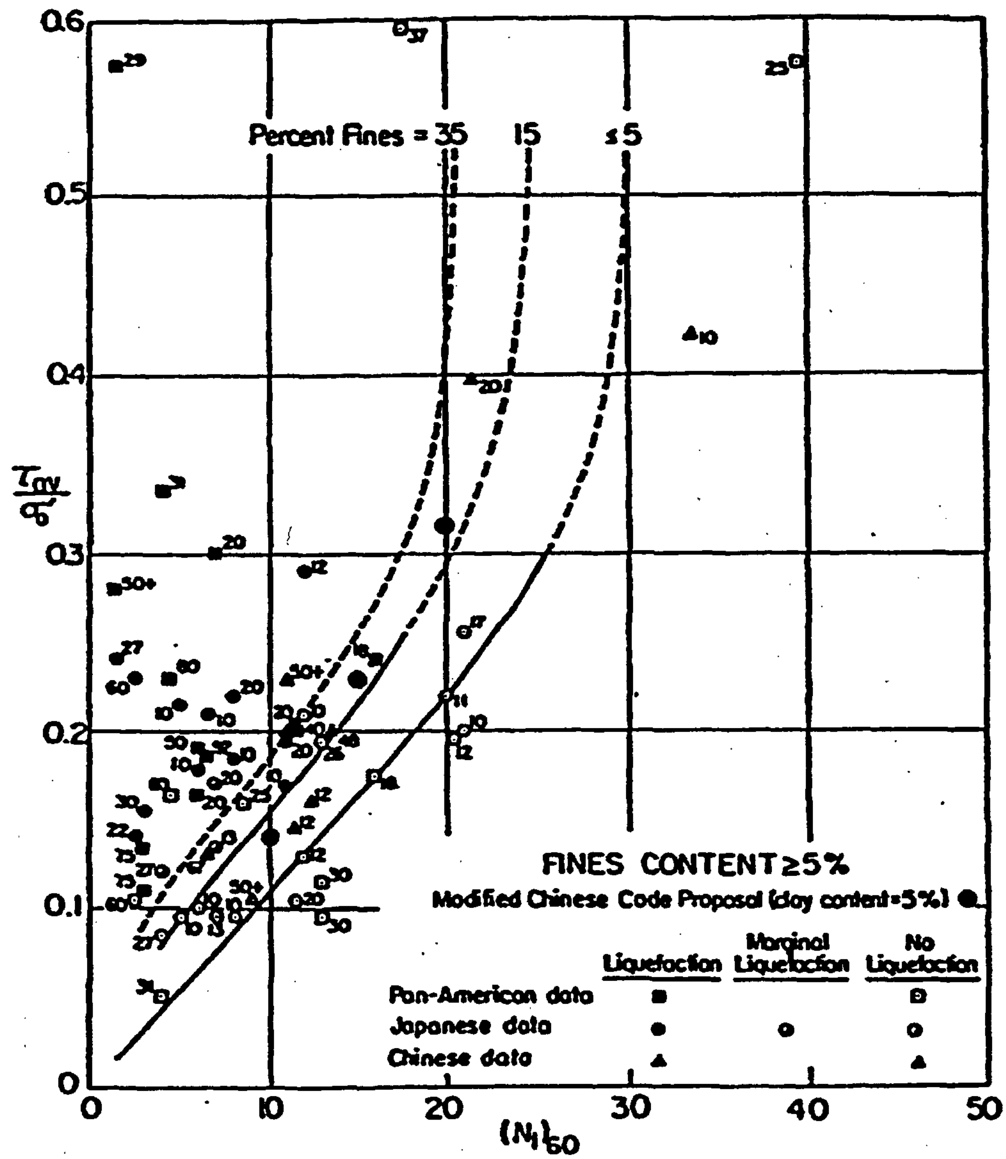


GEOMATRIX

SECTION B-B'
From FSAR

Figure 3

Project No. 1134A

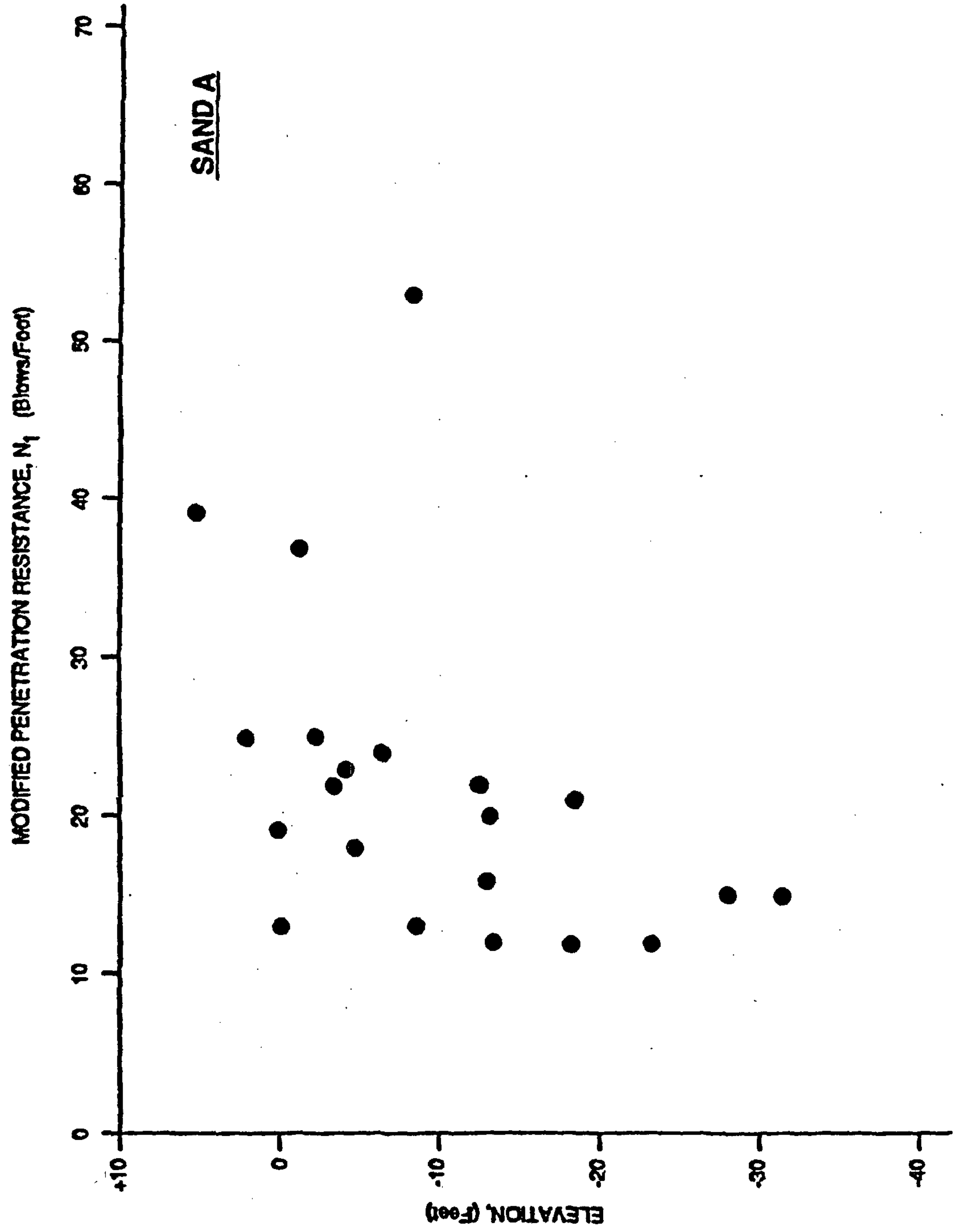


GEOMATRIX

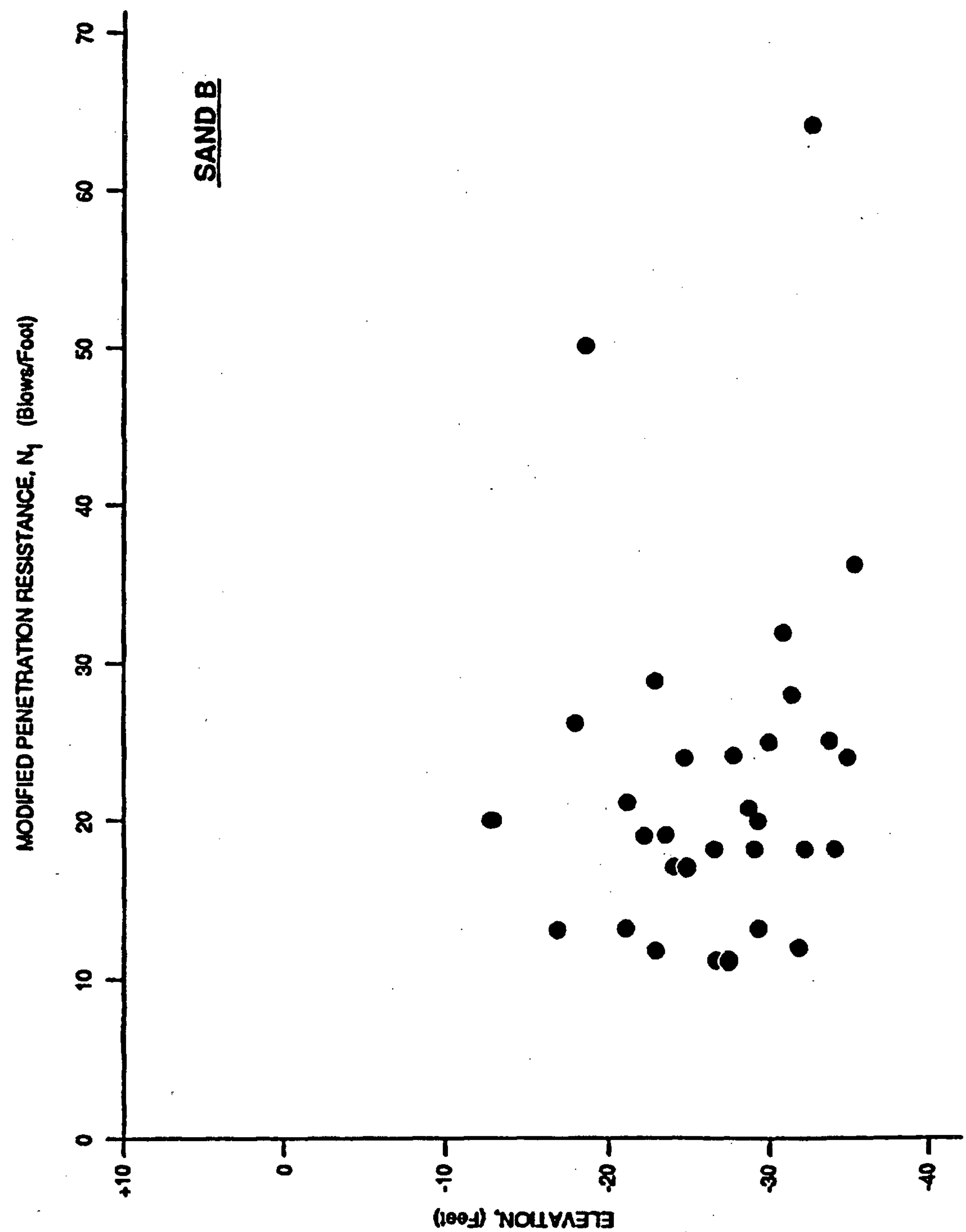
RELATIONSHIP BETWEEN STRESS RATIOS CAUSING LIQUEFACTION AND N_1 VALUES FOR $M = 7\text{-}1/2$ EARTHQUAKES FROM SEED AND OTHERS, 1985

Figure 4

Project No. 1134A



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.

See A-9 where heard believes MELCOR needs to be validated against TMI-2

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.

But the NUREG does discuss The mass/heat for MELCOR and rod inventory

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 and a search also for non-conservatisms?!

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.

interesting actually the mission is discussed in the brochure

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.

The SOARCA Study

Reviewer: Jacquelyn C. Yanch, PhD

What is the Impact on Health of Elevated Radionuclide Levels in the Environment?

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

Summary of Review (One-Page)

Part A: Review Comments

Part B: Our fundamental lack of knowledge about the health impact of the post-accident radiation scenario.

What is our current understanding of the health effects of the radiation conditions represented by the return-home dose-limits? The data used and the process involved in establishing radiation risk estimates and for setting dose limits are discussed.

Part C: Recommendations

Strategies for improving our understanding of radiation effects in the dose regime most relevant to a severe reactor accident are discussed.

Appendix List of Acronyms and Dose Conversion Table

Literature Cited

The SOARCA Study

Reviewer: Jacquelyn C. Yanch, PhD

What is the Impact on Health of Elevated Radionuclide Levels in the Environment?

Review Summary:

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

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

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

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

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

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

Part A: Review Comments

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

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

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

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

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

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

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

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

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

As a society we have developed strategies for dealing with our lack of knowledge of the health effects from low dose-rate radiation. We need these strategies to guide radiation protection policies. For routine radiation protection our limited understanding of the potential hazards presents little difficulty, mostly due to the specifics of this scenario, namely who controls the radiation source, who bears the risk, and who bears the costs of keeping the doses very low. [This is discussed further in Point 5.] Can these strategies also be used to project long-term health effects from accidental exposures? Caution is often expressed *against* extending these strategies to predicting the long term effects of small doses to a large population [eg. 2-4], however, as discussed in the SOARCA documentation, few recommendations for precisely *how* to project the effects of small doses have been provided by agencies or committees involved in generating risk estimates. Therefore, in the absence of a better approach, this caution is routinely ignored by the scientific community in situations where the potential magnitude of the impact of low doses is of interest and the general approach taken in radiation protection is nearly universally employed.

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

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

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

<u>A.4 Level and depth of new knowledge: reactor systems versus health impact.</u>

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

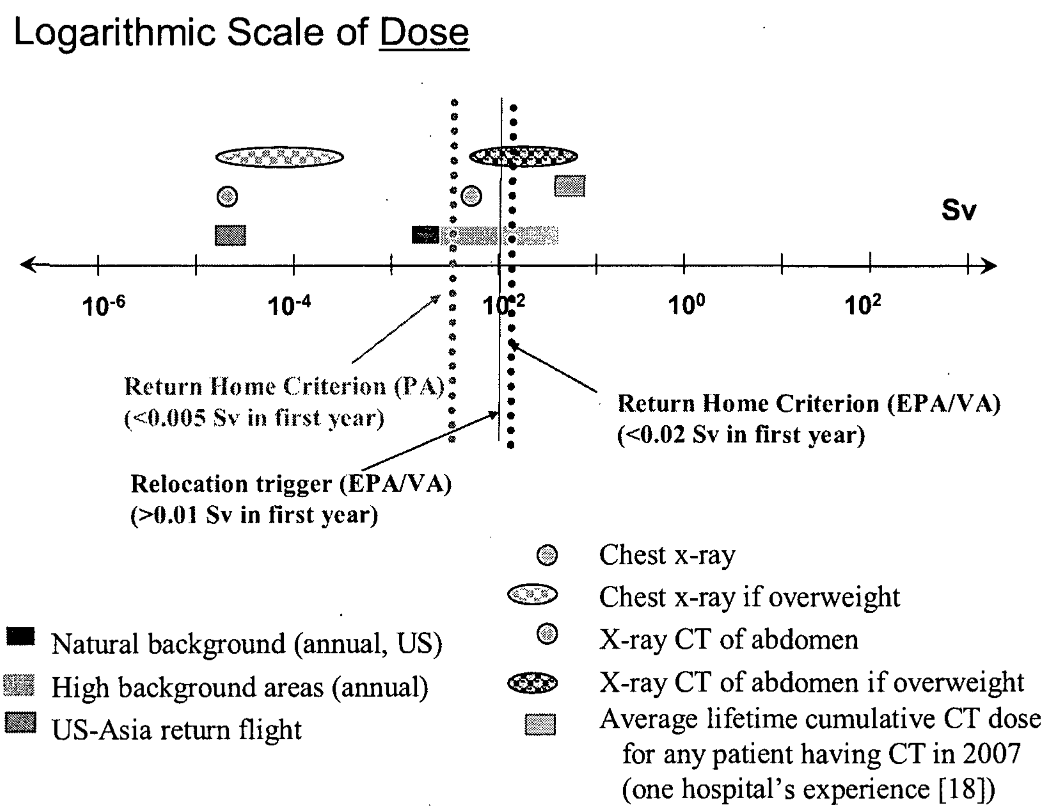

Figure 1. A logarithmic scale of dose showing a range of activities exposing people to ionizing radiation. Shown are annual background dose to residents of the US [14] and to those living in high background regions of the world [15]. Also shown are doses from airline travel and those from radiographic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

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

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

<u>B.1 The A-bomb Survivors Dataset:</u>

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

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

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

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

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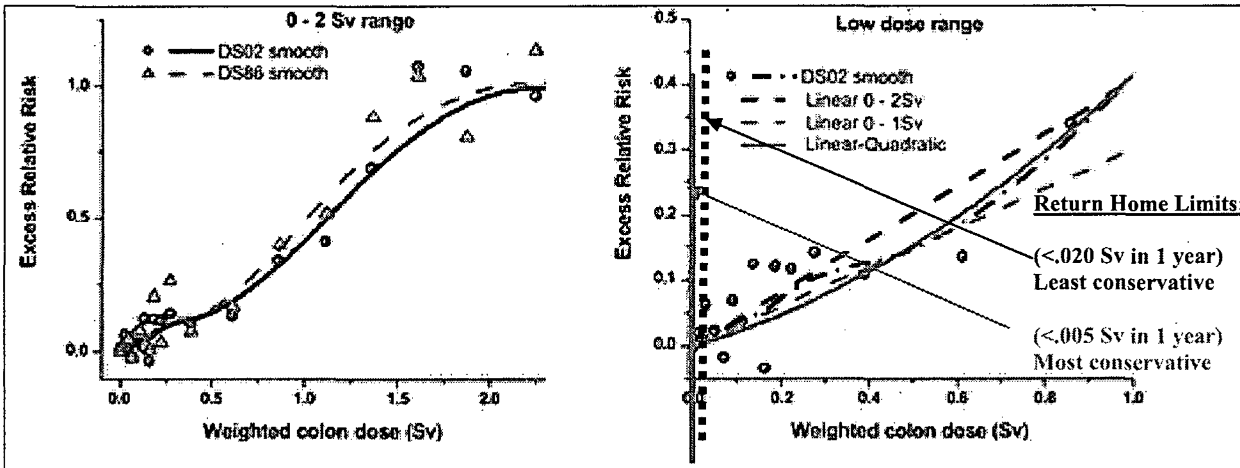
[2] This is the largest and most general population exposed to radiation over a wide range of doses. It is also the largest single cohort of generally healthy individuals exposed to radiation as children.

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

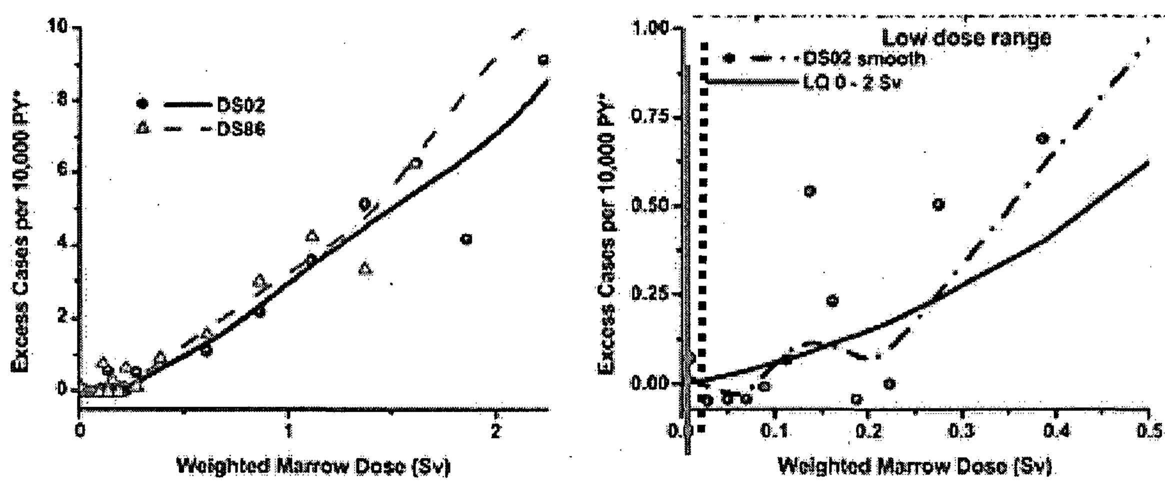
[4] Health effects other than cancer have been examined at high doses; however at low doses non-cancer risks are especially uncertain, according to BEIR VII [7], and are not typically incorporated into risk estimates. Radiation induced mutations in sperm or eggs resulting in heritable disease is so low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

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



(a) Excess relative risk of solid cancer mortality.

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



(c) Excess absolute risk of leukemia mortality.

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

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

In 1(b) and (d) the the RHDLs are superimposed on the low-dose cancer risk plots The vertical lines representing the Pennsylvania (5 mSv) and the EPA RHDL (20 mSv) are the doses accumulated over the first year of return-home habitation. [This is an important distinction from the A-bomb doses referred to on the plot's abscissa which were delivered in less than a minute. See more on this point in Part B.2.]

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

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

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

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

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

- Using a linear extrapolation of the A-bomb data down to low doses we generate an estimate of the risk of cancer fatality per Sievert.
- We determine the radiation dose needed to result in a risk of death that matches the risk of on-the-job fatality for workers in safe industries such as trade, retail, and government; this dose becomes the maximum permissible dose to workers (50 mSv).

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

- We then divide the occupational dose limit by 10 to serve as the dose limit to members of the public (5 mSv).
- We then multiply this by 4 (20 mSv first year RHDL, EPA) or not at all (5 mSv per year RHDL, PA) in response to an unexpected or emergency-related release of radionuclides in the environment.

<u>Thus, the public is not permitted to return to their homes until the risk they are exposed to from residual radiation has dropped to 40% or 10% of the risk a government worker faces from an on-the-job fatality</u>.

The public will be paying a very considerable price (e.g. living for many years away from their home, perhaps permanently, etc.) simply to avoid a risk that we can't confirm exists and even our best estimate predicts that the risk is minor. It is unlikely that members of the public would consider this an acceptable trade-off.

<u>B.2 The accident scenario requires a different approach than for routine radiation protection.</u>

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

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

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

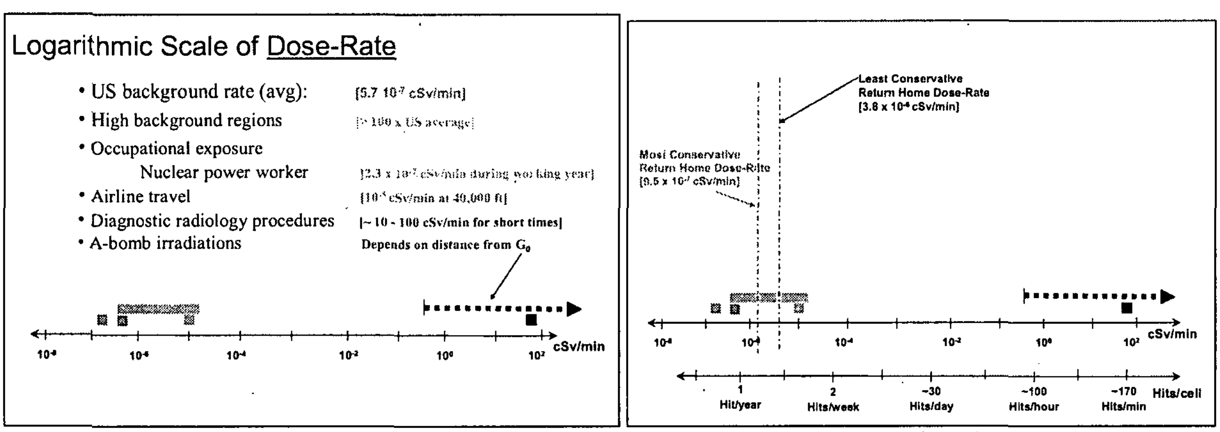

Figure 2. Both (a) and (b) depict the same logarithmic scale of dose-rate. Units are cSv/min (rem/minute). An additional scale is provided on (b) and shows number of radiation tracks (hits) per cell in the body per unit time (assuming 1 hit/cell from an average whole body dose of 3 mSv. Shown in (a)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

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

NCRP 64 "Influence of Dose and its Distribution in Time on Dose-Response Relationships for Low-LET Radiations" (1980) investigated a dose-rate effectiveness factor (DREF) to be used in situations when radiation dose is delivered over time as opposed to delivered acutely [31]. The DREF allows us to take the risk estimates generated using the A-bomb survivor data and apply them to occupational exposures.

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

> "exposure times constituting a significant or sizeable fraction of the life span.... Long enough to permit age-dependent changes in the radiosensitivity of the target (e.g. changes in susceptibility to tumor induction or expression with age)" [31].

On the other hand, the true dose rate effect describes shorter-term exposures not influenced by factors important in protraction effects (i.e. includes effects of DNA repair but not of age) [31].

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

Clear differences in magnitude were noted between PFs and DREFs. Protraction Factors for cancer induction generated by long-term irradiation ranged from 6.6 to 12.8 with a mean of 10. That is, long-term, low dose-rate radiation is approximately 10 times less effective in cancer induction than the same dose delivered acutely. Shorter-term exposures generated an average DREF of 4 (range 1.1 – 10). Thus the magnitude of the dose-rate factor depends on the total length of the irradiation [31].

In summarizing dose-rate effects for radiation-induced tumorigenesis, NCRP 64 recommends a DREF of 2 – 10. Subsequent examination of the dose-rate effect in later NCRP publications and by other committees (ICRP, UNSCEAR, BEIR, EPA) rely heavily on NCRP 64 and usually invoke a conservative DREF of 2 (or 1.5) when converting risk estimates based on acute radiation delivery to the low dose-rate scenario of occupational exposure. Each committee has also reviewed relevant studies of the dose-rate effect published since NCRP 64. Again, however, due in part to the difficulties in actually observing deleterious effects at very low dose-rate, none of the laboratory studies examined (including those cited in BEIF VII [7]) address the dose-rates encountered upon returning home following a severe reactor accident[6].

Figure 3 reproduces the logarithmic dose-rate plot shown in Figure 2(b) and superimposes the dose-rates compared in each of the studies referenced by NCRP 64 in their summary table from which the DREF for tumorigenesis is obtained.

Note that the lowest dose-rate examined in establishing the DREF [31] is still a factor of 100 greater than the least conservative RHDL; most of the "low" dose-rates studied are more than 1000 times greater than the RHDLs. One of the reasons for this is the fact that biological effects at lower dose-rates could not be observed.

[6] Some data regarding repeated delivery of low doses from diagnostic radiology (sometimes referred to as 'fractionation') are used but these remain high dose-rate delivery scenarios, separated in time.

In fact, low doses and low dose-rates lead to *increased* longevity rather than the decreased lifespan seen at higher doses and dose-rates. In addressing the apparent life lengthening at low dose-rates, the NCRP interprets this effect as reflecting "a favorable response to low grade injury leading to some degree of systemic stimulation." They go on to state that "...there appears to be little doubt that mean life span in some animal populations exposed to low level radiation throughout their lifetimes is longer than that of the unirradiated control population." [31, p 104]

Thus, the consequences of the radiation delivery at dose-rates used in estimating the DREF are quite possibly different than the consequences of the much lower dose rates typical of the RHDL.

A DREF of 2 is used in the MACCS2 code and is therefore incorporated into the SOARCA estimates of latent cancer fatalities. However, using the value of 2 for the scenario involving prolonged exposure to elevated radionuclides in the environment is likely not appropriate. For the situation of long term exposure to radiation, such as the situation facing residents who return to their homes and are irradiated by the return-home dose levels, the effect of radiation <u>protraction</u> should be applied. This is because these residents will likely be irradiated for the remainder of their lives (albeit to an ever decreasing extent) from residual ^{137}Cs in the environment. As noted in NCRP 64, very long-term radiation reduces the biological consequences to a greater extent than is predicted when just using the DREF [31].

In choosing to adopt a DREF of 2 rather than a larger number, BEIR V states that the higher values of DREF listed in NCRP 64 reflect situations involving continuous daily irradiation until death, but found that this "may be an unlikely circumstance for humans except as a result of natural background radiation." [2] Residents returning home, post-accident will, in fact, be exposed to continuously daily irradiation, presumably until death, and the impact of dose <u>protraction</u> rather than merely the <u>true dose-rate effect</u> *should* be taken into consideration when RHDLs and relocation triggers are considered.

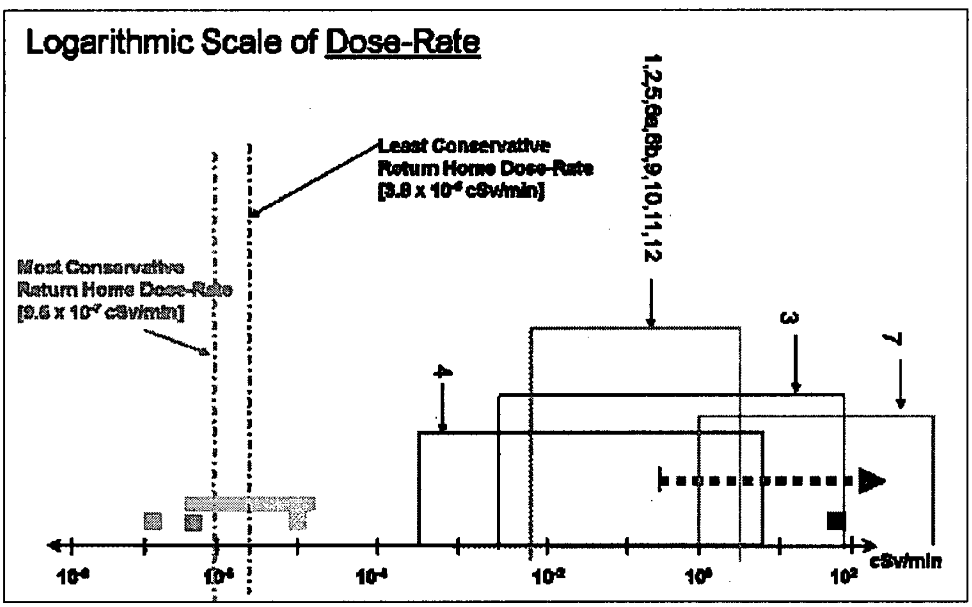

Figure 3. The logarithmic dose-rate chart shown in Figure 2(b) is displayed again here; superimposed are the dose-rate comparisons made by studies referenced by NCRP 64 in estimating dose-rate effectiveness factors (DREF) for tumorigenesis. A total of 12 datasets are cited; the dose-rates compared by each study are indicated, numbered here according to their number in Table 9.3, NCRP 64 [31]. Note that none of the comparisons examined dose rates similar to the RHDLs, or even within a factor of 100.

Thus, extending use of a DREF of 2.0 to situations involving long-term irradiation due to radionuclides in the environment represents a significant overestimate of the detriment. This overestimate, according to the assessment of NCRP 64 [31], is approximately a factor of 5. BEIR VII considers a DREF of 1.5 most 'believable' and cites ICRP 1990 who use a value of 2.0 while "...recognizing that the choice is somewhat arbitrary and may be conservative." <u>Overly-conservative risk estimates have a role to play in routine radiation protection scenarios, but they become unnecessarily burdensome when their use requires important response on the part of individual members of the public</u>.

<u>B.3 Uncertainty in the risk estimates:</u>

Uncertainty analyses of the risk per unit dose estimate generated using the LNT model have been performed by a number of committees and agencies [32,33]. Relative uncertainties about the nominal risk estimates are generally estimated to be in the range 200 to 400 % when constraining the risk versus dose relationship to pass through (0,0). That is, the 90% confidence interval about the nominal risk estimates covers a range of risk estimates that varies by a factor of approximately 7.

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

The largest contribution to uncertainty (38%) comes from estimating the DREF. Figure 4b shows the values of DREF considered in the uncertainty analysis and the relative probabilities of being correct, as subjectively assessed by the NCRP [32] and the EPA [33]. Both agencies assigned a most likely value of 2.0 with the probability of larger values diminishing rapidly. No consideration is made of DREFs as large as the PFs (which are more appropriate to the return home irradiation conditions) but this would *substantially increase the uncertainty* associated with the risk estimates. [Recall the protection factor ranges from 6.8 – 12.8.]

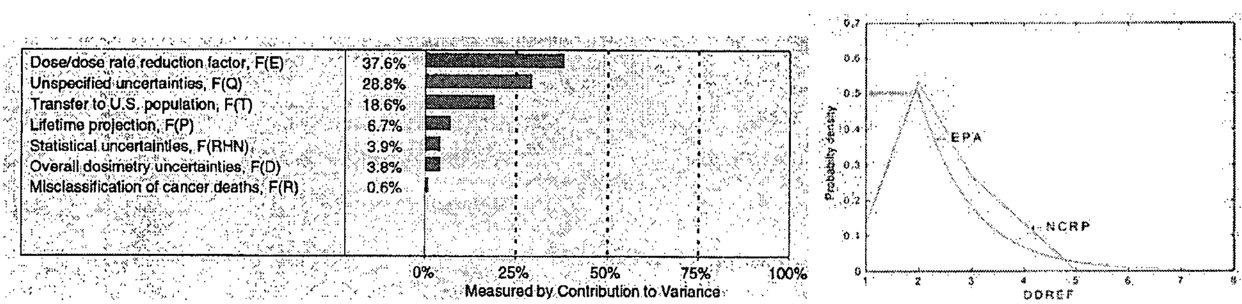

Figure 4. (a) Relative importance of various components of uncertainty on the uncertainty associated with lifetime cancer fatality risk estimates for a general population. Taken from NCRP 126 (p. 72) [32]. (b) [EPA 1999 Addendum] [33]

Assuming a very large DREF (i.e. DREF $\rightarrow \infty$) implies the existence of a threshold dose below which no increase in cancer fatality would be seen. The latest evaluation of the A-bomb survivor data [34] has demonstrated that a threshold dose of 0.04 Sv (4 mSv) fits the data with a statistical significance level *equal* to that observed when using the LNT model[7]. NCRP and EPA clearly state that the <u>'choice of dose-response model' is not considered</u> in determination of the 200-400% uncertainty in low dose-rate risks estimates (see

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[7] *"Based on fitting a series of models with thresholds at the dose cutpoints in the person-year table, the best estimate of a threshold was 0.04 Gy with an upper 90% confidence bound of about 0.085 Gy. However this model did not fit significantly better than a linear model."* from Preston et al, 2007 [34].

Figure 4a). However, if both the LNT and a threshold of 0.04 Sv (4 rem), acutely delivered, fit the data equally well, this surely implies an even greater uncertainty in the risk estimates[8].

In summary, the 200-400% uncertainty in risk estimates derived for radiation protection scenarios is likely to be a very significant underestimate of the uncertainty when applying these risk estimates to the effects of long-term, protracted exposure. In fact, the uncertainty is likely to be at least a factor of 10 and possibly much greater.

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

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

Part C: Recommendations

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

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

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

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

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

A couple of decades ago the doses received from diagnostic radiology were relatively small and experienced by relatively few individuals. Today, however, radiological exams are used for addressing a much broader range of medical questions and are performed on a much bigger fraction of the population. More important is the fact that we've begun to make routine use of the more dose-intensive procedures of x-ray computed tomography (CT) and interventional fluoroscopy [17]. The result is that the average US resident receives as much radiation dose from diagnostic radiology procedures as from all natural background radiation sources, combined.

Thus, on a routine basis, and for a variety of reasons, we deliberately and carefully irradiate most members of the U.S. population, exposing them to a wide range of doses depending on the reason for the exam, the part of the body being imaged, and the patient's body thickness. It makes sense to maintain a registry of radiation doses for everyone irradiated. This registry would not be a "de-identified" patient radiation dose data-base (as proposed recently by the FDA as a starting point for establishing consistent exam parameters across medical institutions [36]), but a registry that allows tracking of an individual's dose over time and, ultimately, for correlation of dose with disease or health status many years later.

Initiating and maintaining a database of patient doses, if done correctly, would provide the single greatest database for low dose radiation exposures. It would also present important advantages for risk determination not available with the LSS study. First, it is unlikely the A-bomb survivor data will ever be able to provide a statistically significant determination of the risk of radiation-induced cancer as a function of dose in the low dose range. Stratification of risk estimates to ask questions about radiosensitivities based on age, gender, medical status, radiation history, etc., will also never be possible in the low dose range with the LSS. On the other hand, hundreds of millions of radiological exams are performed each year in the U.S. on people of all ages. While some radiological procedures are performed on patients with an underlying medical condition that could represent a confounding factor for any future analysis of radiation-induced health consequences, many procedures are performed in cases of acute trauma or in other situations with "no evidence of disease". Even if only a subset of the radiological exams were appropriate for long-term evaluation of the effects of radiation on health we will still quickly accumulate a sufficient sample size for the statistical power we need to answer the question: what impact do low doses of radiation have on our health? With over 350 million diagnostic radiology or nuclear medicine exams performed in the U.S. in 2006 alone [17] the statistical precision possible is very quickly greater than that with the A-bomb survivor study (<98,000 irradiated in the LSS), and we will be able to stratify the data so that we can assess the impact of low dose radiation on potentially sensitive subgroups within the population.

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

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

For instance, while we do not currently record the doses received by individual patients, strategies do exist for determining individual organ doses for each patient from each procedure[9] and such information could be stored in the patient's electronic record. Other data naturally included in the electronic medical record and of potential relevance to radiation response would be medications at the time of the exam, antidepressants, diet, and prior radiation history.

Analysis of a medical radiation dataset would provide a valuable supplement to the LSS data at <u>low doses</u>, essentially the only dose range of interest in the post-accident scenario. It would not, however, provide direct

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

information about the effects of low dose-rate radiation. [Diagnostic radiology represents high dose-rate delivery, perhaps repeated (e.g. a week or a decade later) to the same or a different part of the body. Given the range of time scales relevant to human biological processes, this situation is unlikely to generate the same biological effect as the same dose spread out continually over time.]

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

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

A concerted approach involving co-ordination of efforts across several countries and involving cohort or case-control studies could be undertaken [15,39]. Use of common study protocols and dosimetry methods will improve the robustness of the data and allow data pooling to increase statistical power. A great deal of information relevant to the issue of triggers for relocation and return-home could be obtained from a thorough and long-standing examination of these populations.

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

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

1. Dose-rates in HBRA span a considerable range. It would be possible to examine health consequences of a full range of dose-rates including those we are now using to trigger relocation and return-home, as well as dose-rates considerably higher. In addition to examining health of the general population in a HBRA, it would be possible to investigate impact on sub-groups within the population. Analysis of the A-bomb survivor data indicate that women are somewhat more sensitive than men to radiation-induced cancers, and that children are substantially more sensitive than adults with the sensitivity changing remarkably with age at time of exposure [41]. The deeper question is: are the same variations in radiosensitivity expected at low dose-rates and to small doses?

2. The wide range of genetic make-up represented by those living in HBRA around the world provides the opportunity to examine the range of genetic susceptibilities to radiation-induced effects by comparing each group living in an HBRA with its own control group (a similar population but living with lower background levels).

3. There may be particular diets, medications or even lifestyles that affect radiation sensitivity. For instance, Lemon et al have shown that including a mix of anti-oxidants in the diet of mice results both in increased longevity and reduced DNA damage (a 6-fold reduction in chromosomal aberrations) following a

single acute dose of irradiation [42]. The neurotransmitter serotonin has been shown to impact the ability of irradiated cells to transmit information about the radiation event to neighboring cells [43]. Would such chemicals as antidepressants, for example, affect the body's response to low dose-rate radiation? And similarly, are there mitigators (medicine) we should be taking when living in an elevated background? Much effort is underway to develop mitigators to redress the effects of high dose/high dose-rate radiation resulting from terrorist activities. Similar efforts could be undertaken to determine if mitigators are useful or necessary when living in slightly elevated backgrounds levels.

4. Given the dearth of data on irradiated humans, animal data are often used to extend our understanding of radiation-induced health effects. However, not all animals show similar responses (as a function of time or of dose) as humans and thus it is difficult to know when and how to extrapolate from animal to human data [44]. Examination of the indigenous animal life, comparing those in the control areas with those in the HBRA, may give us the opportunity to study the magnitude of species-to-species or strain-to-strain differences in response to radiation, perhaps shedding light on the degree to which we can rely on animal data to inform our understanding of human radiobiology.

5. Much of the increase in radiation dose in a HBRA comes externally from elevated gamma-emitting radionuclides in the soil, similar to the irradiation route envisioned for the post-accident return-home scenario. In some areas, however, substantial increases in internal radionuclides contribute to elevated dose through food and water consumption. How different are the effects of higher organ doses when the dose is delivered via external gamma rays than when the elevated dose results from eating food elevated in ^{137}Cs or other radionuclides? In still other HBRAs, the increased radiation is due to higher elevations where contributions to dose from cosmic radiations are increased [38]. These radiations have higher LET but the impact of low dose-rate high LET radiation is not currently well understood.

6. The number of people living in some HBRAs is very large, potentially providing the necessary statistical power to generate statistically significant estimates of the differences in health impact of elevated background radiation. Over 20 million people live in Mexico City (elevation 2240 m) where the cosmic radiation dose is more than double the world average [38]. More than 350,000 residents of the Kerala area of India are currently being studies [39] via interviews to examine factors associated with lifestyle, medications, etc., and dosimetry has been performed in and around over 70,000 homes. Over 125,000 residents of the Guangdong Province in China have been under study since 1987 [45].

7. Is an increase in our natural cancer rate the *only* thing we need to worry about? Data arising from analyses of the A-bomb survivors suggest that other health effects may be elevated (although the data are insufficient for detailed investigation. Data from animal studies suggest the relative causes of death as a function of age change after exposure to radiation [46].

C3. Summary:

Both strategies proposed here for developing an improved understanding of the health impact of low dose or low dose-rate radiation will require a very significant effort to generate the data we need to develop our understanding of the real consequences of a severe reactor accident. However there is considerable on-going effort to take advantage of. For instance, experience with life-long tracking of radiation dose is available with radiation registries used to track occupational radiation doses [35], and of course, extensive high quality analysis of long-term health effects from radiation exposure as a function of age and gender has been performed for decades by the Radiation Effects Research Foundation in their analysis of the participants in the Life Span Study [27,34]. This existing experience would represent a valuable starting point, for instance, for a registry of individual patient doses.

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

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

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

The risk estimates from NUREG 6555 are used, after modification by a DREF of 2.0, in the quantification of latent cancer fatality risk in the SOARCA study. Questions in NUREG-6555 were put to 13 experts and deal with cancer incidence and cancer death rates for a given radiation dose. For all but one case involving exposure to low LET radiation, the experts were asked to provide their estimate of the risk of fatal cancer resulting from high dose, high dose-rate radiation (1 Gy delivered over 1 minute).

However, of much greater use would be expert estimates of the effects of low dose-rate and low dose gamma radiation, i.e. conditions identical to those we will encounter upon returning home following a reactor accident. [An abundance of recent data relevant to these questions comes from studies of life-long irradiation of small animals, and radiobiological examination of low dose and low dose-rate radiation on cells and tissues in the laboratory (e.g. impact of doses below 5 mGy [48-50]. The field of radiobiology has undergone rapid changes in the last decade with advancements in biological assays and interrogation methods that make it possible to begin addressing biological responses at lower doses than possible in decades past.] Expert opinion should be solicited regarding the risks associated with long-term, protracted radiation exposures of the public. Such risk estimates will be far more useful in generating a 'state-of-the-art' estimate of the consequences of elevated radionuclides in the environment.

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

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

If, in the event of an accident, we are to make hard decisions involving trading increased exposure to radiation against abandoning homes, public buildings, major industries, destroying livestock and crops, then we need some solid data on which to base these decisions. These data do not exist, and, further, to generate these data will require a wait-time of decades. Therefore, in order to generate data we need to begin now.

Appendix

Glossary of Acronyms:

RHDL	Return home dose limits
NAS BEIR	National Academies of Sciences Biological Effects of Ionizing Radiation (US)
NCRP	National Commission on Radiological Protection (US)
ICRP	International Commission on Radiological Protection
EPA	Environmental Protection Agency (US)
UNSCEAR	United Nations Scientific Committee on the Effects of Atomic radiation
DS02	Dosimetry Study 2002
ALARA	As Low as Reasonably Achievable
LSS	Life Span Study (A-bomb survivor dataset)
LET	Linear Energy Transfer
HBRA	High Background Radiation Area

Units of Dose (and effective dose equivalent):

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

Literature Cited:

1. Manual of Protective Action Guides and Protective Actions for Nuclear Incidents, Office of Radiation Programs, US Environmental Protection Agency, Washington DC, May 1992.

2. "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation: 1980" Committee on the Biological Effects of Ionizing Radiations, National Academy Press, Washington, DC, 1980.

3. "Radiation Risk in Perspective" Position Statement of the Health Physics Society, adopted January 1996, reaffirmed March 2001 and revised August 2004. http://www.hps.org

4. ICRP Report 103: The 2007 Recommendations of the International Commission on Radiation Protection, 2007.

5. "Probabilistic Accident Consequence Uncertainty Analysis, Late Health Effects Uncertainty Assessment" NUREG/CR-6555, EUR 16774, Prepared for Division of Systems Technology, US Nuclear Regulatory Commission and Commission of the European Communities, 1997.

6. "Health Effects of Exposure to Low Levels of Ionizing Radiation, BEIR V" National Research Council, National Academy Press, Washington, DC, 1990.

7. "Health Risks from Exposure to Low Levels of Ionizing Radiation, BEIR VII Phase 2" Committee to Assess Health Risks from Exposure to Low Levels of Ionizing Radiation, Board on Radiation Effects Research, Div. on Earth and Life Studies, National Research Council of the National Academies, National Academies Press, Washington, DC, 2006.

8. "Effects of Ionizing Radiation: United Nations Scientific Committee on the Effects of Atomic Radiation – UNSCEAR 2006 Report, Volume 1 – Report to the General Assembly, with Scientific Annexes A and B, United Nations Office at Vienna, July 2008.

9. "Evaluation of the Linear-Nonthreshold Dose-Response Model for Ionizing Radiation" NCRP Report No. 136, National Council on Radiation Protection and Measurement, Bethesda, MD, June 4, 2001.

10. "Estimating Radiogenic Cancer Risks" EPA 402-R-93-076, U.S. Environmental Protection Agency, Washington DC, June 1994.

11. D.C. Aldrich, et al "Technical Guidance for Siting Criteria Development" NUREG/CR-2239, SAND-81-1549, Sandia National Lab, Albuquerque, NM, Dec. 1982.

12. B.W. Emmerson "Intervening for the protection of the public following a nuclear accident" *IAEA Bulletin*, 3, 12-18, 1988.

13. M. Crick "Nuclear and radiation safety: Guidance for emergency response" *IAEA Bulletin*, 1, 23-27, 1996.

14. "Ionizing Radiation Exposure of the Population of the United States" NCRP Report No. 160, March 3, 2009, National Council on Radiation Protection and Measurement, Bethesda, MD.

15. "High Levels of Natural Radiation and Radon Areas: Radiation Dose and Health Effects" Proceedings of the 6th International Conference on High levels of Natural Radiation and Radon Areas, Osaka, Japan, Sept. 6-10, 2004, Elsevier 2005.

16. F.A. Mettler, W. Huda, T.T. Yoshizumi and M. Mahesh "Effective Doses in Radiology and Diagnostic Nuclear Medicine: A Catalog" *Radiology*, 248, 254-263, 2008.

17. F.A. Mettler, et al "Radiologic and Nuclear Medicine Studies in the United States and Worldwide: Frequency, Radiation Dose, and Comparison with Other Radiation Sources, 1950-2007" *Radiology*, 253(2), 520-531, 2009.

18. A. Sodickson, et al "Recurrent CT, Cumulative Radiation Exposure, and Associated Radiation-induced Cancer Risks from CT of Adults" *Radiology*, 251(1), 175-184, 2009.

19. Background for Protective Action Recommendations: Accidental Radioactive Contamination of Food and Animal Feeds" HHS Publication FDA 82-8196, US Dept. of Health and Human Services, Bureau of Radiological Health, Rockville, MD 20857, August 1982.

20. International Commission on Radiological Protection, ICRP Report No. 23: Report of the Task Group on Reference Man, Pergamon, Oxford, UK, 1975.

21. K.M. Flegal, M.D. Carroll, R.J. Kuzmarski and C.L. Johnson "Overweight and obesity in the United States: prevalence and trends, 1960-1994" *Int J Obes Relat Metab Disord*, 22(1), 39-47, 1998.

22. J.C. Yanch, R.H. Behrman, M.J. Hendricks and J.H. McCall "Increased Radiation Dose to Overweight and Obese Patients from Radiographic Examinations" *Radiology*, 252(1), 128-139, 2009.

23. T. Imanaka and H. Hiromi "Investigation continues on the atomic bomb dosimetry" *Radiat. Envir. Biophys.*, 47, 297-299, 2008.

24. H.M. Cullings et al "Dose Estimation for Atomic Bomb Survivor Studies: Its Evolution and Present Status" *Radiation Research*, 166, 219-254, 2006.

25. K. Takamiya et al "Nickel-63 production in copper samples exposed to the Hiroshima atomic bomb: estimation based on an excitation function obtained by neutron irradiation experiments" *Radiat. Environ. Biophys.*, 47, 343-348, 2008.

26. M. Hoshi "Intercomparison study on ^{152}Eu gamma ray and ^{36}Cl AMS measurements for development of the new Hiroshima-Nagaski Atomic Bomb Dosimetry System 2002 (DS02)" *Radiat. Environ. Biophys.*, 47, 313-322, 2008.

27. D.L. Preston et al "Effect of Recent Changes in Atomic Bomb Survivor Dosimetry on Cancer Mortality Risk Estimates" *Radiation Research*, 162, 377-389, 2004.

28. "Limitations of Exposure to Ionizing Radiation" NCRP Report No. 116, March 31, 1993, National Council on Radiation Protection and Measurement, Bethesda, MD.

29. Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities, 2008 Forty-First Annual Report, http://www.reirs.com/nureg2008/nureg2008.pdf, accessed 20 May 2010.

30. L.E. Feinendegen and R.D. Neumann "Physics Must Join with Biology in Better Assessing Risk from Low-Dose Irradiation" *Radiation Protection Dosimetry*, 117(4), 346-356, 2005.

31. "Influence of Dose and its Distribution in Time on Dose-Response Relationships for Low-LET Radiations" NCRP Report No. 64, April 1, 1980, National Council on Radiation Protection and Measurement, Bethesda, MD.

32. "Uncertainties in Fatal Cancer Risk Estimates Use in Radiation Protection" NCRP Report No. 126, Oct. 17, 1997, National Council on Radiation Protection and Measurement, Bethesda, MD.

33. "Estimating Radiogenic Cancer Risks/Addendum: Uncertainty Analysis" EPA 402-R-99-003, U.S. Environmental Protection Agency, Washington DC, May 1999.

34. D.L. Preston et al "Solid Cancer Evidence in Atomic Bomb Survivors: 1958-1998" *Radiation Research*, 168, 1-64, 2007.

35. K. Shrader-Frechette "Trimming Exposure Data, Putting Radiation Workers at Risk: Improving Disclosure and Consent Through a National Dose-Registry" *American Journal of Public Health*, 97(9), 1782-1786, 2007.

36. "Radiation-emitting Products, White Paper: Initiative to Reduce Unnecessary Radiation Exposure from Medical Imaging" FDA/ U.S. Food and Drug Administration, February 2010.

37. C. Dissanayake "Of Stones and Health: Medical Geology in Sri Lanka" *Science*, 309(5736), 883-885, 2005.

38. M. Durante and L. Manti "Human response to high-background radiation environments on Earth and in space" *Advances in Space Research*, 42, 999-1007, 2008.

39. E. Cardis "Commentary on information that can be drawn from studies of areas with high levels of natural radiation", International Congress Series 1276, 118-123, 2005.

40. "Effects of Ionizing Radiation: United Nations Scientific Committee on the Effects of Atomic Radiation – UNSCEAR 2000 Report, Volume 1 – Report to the General Assembly, with Scientific Annexes A and B, 2000.

42. J. A. Lemon, C. D. Rollo and D. R. Boreham, "Elevated DNA damage in a mouse model of oxidative stress: impacts of ionizing radiation and a protective dietary supplement," Mutagenesis 2008 23(6):473-482

43. C. Mothersill, R. Saroya, R. W. Smith, H. Singh, and C.B. Seymour "Serum Serotonin Levels Determine the Magnitude and Type of Bystander Effects in Medium Transfer Experiments" Radiation Research, published on-line ahead of pint: http://www.rrjournal.org/doi/abs/10.1667/RR2036.1

45. W. Luxin et al "Epidemiological Investigation of Radiological Effects in High Background Radiation Areas of Yangjiang, China" *J. Radiat. Res.*, 31, 119-136, 1990.

46. B.A. Carnes, N. Gavrilova and D. Grahn "Pathology Effects at Radiation Doses below those Causing Increasing Mortality" *Radiation Research*, 158, 187-194, 2002.

47. N. Pauwels, F. Hardeman and K. Soudan "Radiological Protective Measures in Highly Industrialized Areas: Do the Existing Intervention Levels Apply?" *Health Physics*, 77(6), 646-653, 1999.

48. T.K. Day et al "Extremely Low Priming Doses of X Radiation Induce an Adaptive Response for Chromosomal Inversions in pKZ1 Mouse Prostate" Radiation Research, 166, 757-766, 2006.

49. E.J. Broome, D.L Browne and R.E.J. Mitchel "Dose Responses for Adaptation to Low Doses of 60Co Rays and 3H Particles in Normal Human Fibroblasts" Radiation Research, 158, 181-186, 2002.

50. R.E.J. Mitchel, P. Burchart and H. Wyatt "A Lower Dose threshold for the In Vivo Protective Adaptive Response to Radiation. Tumorigenesis in Chronically Exposed Normal and Trp53 Heterozygous C57BL/6 Mice" Radiation Research, 170, 765-775, 2006.

The SOARCA Study

What is the Impact on Health of Radionuclides in the Environment?

Reviewer: Jacquelyn C. Yanch, PhD

Abstract:

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

So what! again so what

U.S.

DOE is throwing money at this

agree, but given where we are going with the issue it's not up to SOARCA or the NRC

I Summary of Review (One-Page)

II Review Comments

III Supporting Material:

Appendix A **Our fundamental lack of knowledge about the health impact of the post-accident radiation scenario.**

The data used and the process involved in establishing radiation risk estimates and for setting dose limits are discussed. Emphasis is placed on illustrating how little we know about the health effects of low dose, and more importantly, low dose-rate radiation.

DHS guidelines give 19 issues that should be considered in the optimization process - rad exposure is only one -

Appendix B **Recommendations**

Suggestions for strategies that would improve our understanding of radiation effects in the dose regime most relevant to a severe reactor accident are provided.

Appendix C **List of Acronyms and Dose Conversion Table**

IV Literature Cited

But as a point of departure, they note EPA suggested C/S plus C/B in Rem and NRC release criterion (25 mrem + ALARA)

The SOARCA Study
What is the Impact on Health of Elevated Levels of Radionuclides in the Environment?

Reviewer: Jacquelyn C. Yanch, PhD

I. Summary:

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

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

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

The fact that we report dose-response model results reflects the lack of agreement in the international community.

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

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

6. The return-home dose limits (set by individual states) are set very low, exacerbating the burden on the public. Even the least conservative return home dose limit is *lower* than the natural background doses in many areas of the world. The criterion used in PA is less than a factor of 2 higher than the average background in the Unites State and is not even one-half the dose received from a single CT exam of the abdomen. In this context, major dose-avoidance strategies such as long-term residential relocation until the return-home dose limit can be met, are unlikely to be in the best interests of the public.

II. Review Comments

1. Dose to the public results from returning home:

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

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

we call it the habitability criterion — why coin a new term.

Getting to the low dose-rate stipulated by the return-home dose-limits (RHDLs) requires either time (for physical decay of the radionuclides and for the effects of weathering) during which residents are living away from their homes, or decontamination procedures (such as scrubbing and/or flushing surfaces; soaking, plowing or removing soil; and removal and replacement of surfaces, etc.). The cost to decontaminate can be considerable; if decontamination costs are greater than the cost of the land or dwelling then the land is considered condemned. [If land is condemned, no dose is accrued by the resident s because they never return].

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

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

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

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

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

As a society we have developed strategies for dealing with our lack of knowledge of the health effects from low dose-rate radiation. We need these strategies to guide radiation protection policies. For routine radiation protection our limited understanding of the potential hazards presents little difficulty, mostly due to the specifics of this scenario, namely who controls the radiation source, who bears the risk, and who bears the costs of keeping the doses very low. [This is discussed further in Point 5.] Can these strategies also be used to project long-term health effects from accidental exposures? Caution is often expressed *against* extending these strategies to predicting the long term effects of small doses to a large population [ ], however, as discussed in the SOARCA documentation, few recommendations for precisely *how* to project the effects of small doses have been provided by agencies or committees involved in generating risk estiamtes. Therefore, in the absence of a better approach, this caution is routinely ignored in situations where the potential magnitude of the impact of low doses is of interest and the general approach taken in radiation protection is nearly universally employed.

Two health consequences of elevated radionuclide levels in the environment are considered in the SOARCA study: (i) early deaths due to the acute radiation syndrome, and (ii) latent cancer fatalities (occurring many years later). Given the expected rate of progression of the various accident scenarios and the anticipated success of evacuation plans, the risk of acute fatalities (which will only occur following very large radiation doses) is either zero or very, very low (conditional risk of 10^{-xx} for a STSBO). Radiation-related risks then become latent cancer fatalities resulting primarily from exposure to the long-term, chronic radiation dose-rates encountered upon being allowed to return home.

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

NO, IS only believe as SOARCA documentation so states - Perhaps not forcefully enough-

NO response from 9 am

4 dose-response models

This strategy for determining the risks of exposure to anthropogenic radiation (assuming a threshold for acute effects, integrating the dose over a 50 year period, assuming cancer is the only impact on long-term health, the use of a DREF of 2.0, and the application of a common risk factor throughout the entire dose range) is broadly consistent with the approach taken by several national and international agencies and committees including BEIR, ICRP, NCRP,

not for the threshold models.

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

UNSCEAR, and the EPA []. Thus the approach taken in the SOARCA study for estimating the impact on health of elevated radionuclide levels in the environment has been performed using a state-of-the-art approach.

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

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

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

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

Not really, though that is a big part

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

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

The state of Pennsylvania sets the RHDL at the same dose-rate used to limit dose to the general public from anthropogenic radiation sources in routine radiation protection, 5 mSv (500 mrem) per year. [Information on the setting of dose limits is provided in Appendix A.] Virginia, My understanding is that, at the time those doses were set, they believed that at the dose was <2 Rem in the first year, it would likely to decay to <500 mRem in the second year. Whether that's still true when the short term dose is now so low, is unknown

which follows EPA guidelines, sets its RHDL a factor of 4 higher for the first year but thereafter matches the 5 mSv/year dose-rate limit used in radiation protection.

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

Certainly DHS's position and why they suggest local optimization

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

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

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

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

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

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

For the evacuated public returning home following a severe reactor accident, the doses received during their first year home are also indicated on Figure 1. [The dose to trigger relocation following an accident at the Surry plant used in the SOARCA study (10 mSv) is also shown.] The bases on which the RHDLs are set are not entirely clear. The FDA has suggested use of 2 standard deviations in natural radiation dose as an acceptable radiation risk. In examining "acceptable" risk the EPA compares risks associated with actions already undertaken and accepted by society. However if the acceptability of risk criterion is to be used we must keep in mind that even the least conservative RHDL (20 mSv in the first year) is lower than the natural background doses in many areas in the world. The RHDL for Pennsylvania (5 mSv) is less than a factor of 2 higher than the average background in the US. The dose accumulated from living the first year under RHDL conditions is not even half the dose measured from *a single CT exam of the abdomen* (11 mSv).

really?

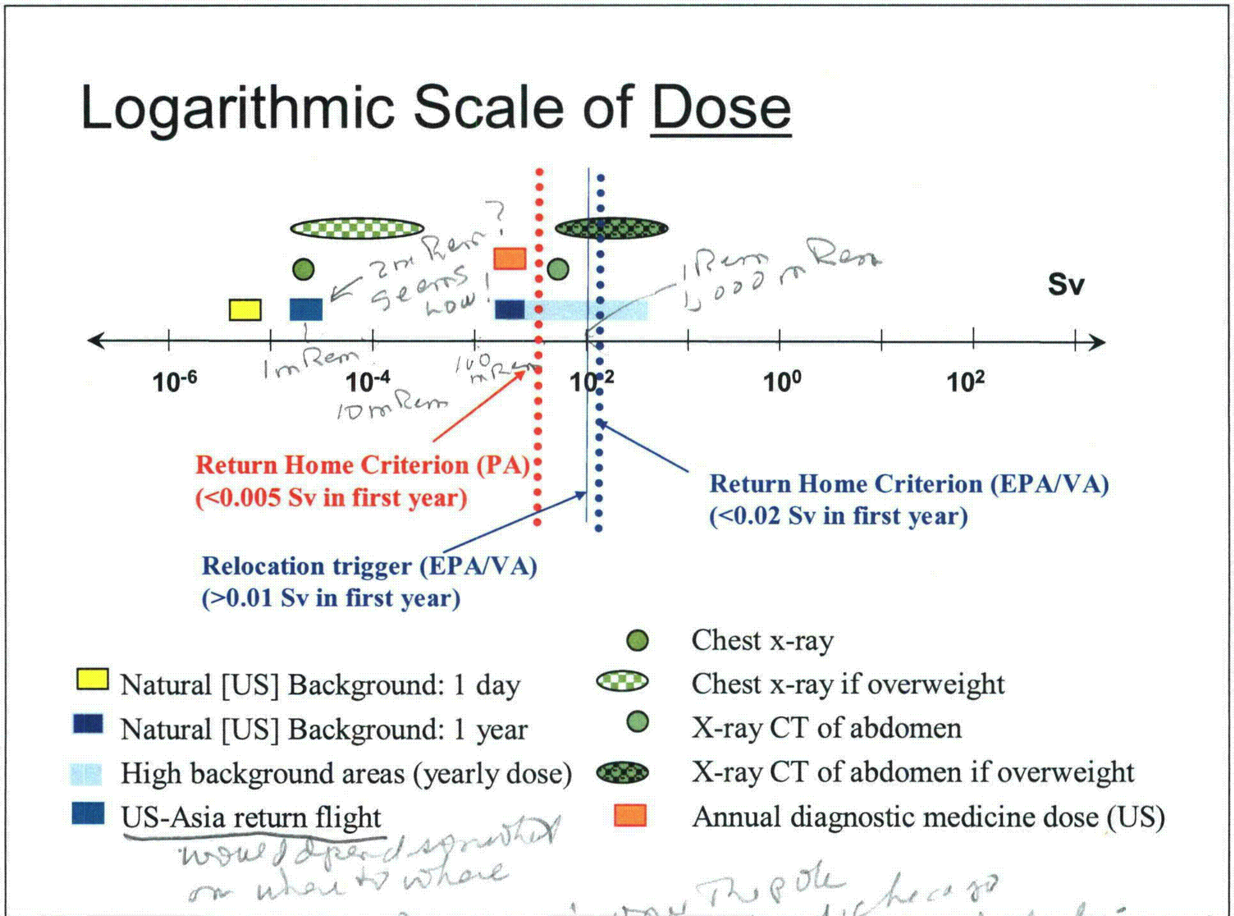
The Superfund folks use 10^{-6} don't they

One rationale the EPA gives for setting the RHDL at 20 mSv is that limiting dose to this level is *reasonably achievable*. It is clear that undertaking the dose avoidance strategies described above will be effective in minimizing dose to the public and thus meeting the dose limit of 20 mSv is achievable. Whether or not it is *reasonable* for the public to undertake these dose avoidance strategies depends on whether they are avoiding a real and significant hazard in doing so. Determining the answer to this question should be a high priority; suggestions for how to generate answers are given in Appendix B.

or 10^{-4} as an upper limit increase in cancer fatality over a lifetime

The EPA drinking water pathway restriction 4 mrem/yr!

10^{-4} they equate with 15 mrem/yr!



Chicago to Beijing is over the pole higher so but Shanghai is over the Aleutian Islands: Both in excess, even,

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

2 m Rem seems to be too low - chart from DOE Office of Science gives NY-London round trip as 10 m Rem

Supporting Material

Appendix A: We do not know what impact long-term exposure to low dose, low-dose-rate radiation will have on human health.

A.1 The A-bomb Survivors Dataset:

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

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

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

Individual dose determination for each LSS participant began with a personal interview to determine precise location and body orientation at the time of the blast. Radiation transport calculations then track neutron and gamma fluence from the fireball through various shielding structures between the individual and ground zero. These calculations have improved dramatically over the past few decades as a result of greater sophistication in the modeling, newly available interaction cross-section data for important isotopes, finer energy group structure for particle transport, and increased confidence in the dosimetry models resulting from detailed comparisons between model predictions and measured data. Photon fluence estimates at varying distances from ground zero have been compared with thermoluminescence measurements in tiles and bricks that were exposed to gamma-rays from the bomb; neutron fluence calculations have been compared with neutron-induced radioactivity in tile, granite,

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

concrete, and soil samples[3]. The ability to compare model-driven fluence estimates with actual measurements taken in various locations and distances from ground zero has resulted in iterative refinement of neutron and photon dose estimates. The doses that individual survivors experienced are thus now known to a good level of accuracy.

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

50 Rem 10 Rem

The dose axes span a very large range (for instance, 2.0 Sv is the equivalent of ~800 times the average natural background dose in the U.S. but delivered within one minute). At large doses (>0.5 Sv) it is clear that survivors are at an increased risk of dying of cancer in later life. At the lowest doses (below ~ 0.1 Sv), the large 'natural' cancer rate in both the exposed and the unexposed populations makes it impossible to declare, with any certainty, what effect small radiation doses have on the cancer fatality rate in exposed persons. But it is precisely this low dose range that we are most interested in. And, since this is the best population available for examining radiation-related health effects, we must use these data in some way to provide guidance for the safe conduct of human activities involving the use or production of ionizing radiation.

yes, based on epi studies - will always have to rely on other data - with the associated uncertainty

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

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

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

[4] Health effects other than cancer have been examined at high doses; however at low doses non-cancer risks are especially uncertain, according to BEIR VII [ ], and are not typically incorporated into risk estimates.

5 Rem now 2 Rem in Europe

determined (on-the-job accidents in retail, government, manufacturing, etc.) and the dose limit is set accordingly.[5] This limit is set at 50 mSv per year.

From Risk Estimate to Public Dose Limits: Dose limits recommended for the worker are reduced by a factor of 10 for application to protection of the public. There are several justifications for setting the limits lower than those for radiation workers. First, the public does not directly benefit (in a wage-related manner) from exposure to the anthropogenic radiation. Second, a wider range of sensitivities is expected to be found in the general population than the adult worker population, and third, the period of exposure can potentially be longer. Dose limits to the public are typically comparable to local variations in natural background radiation (Crick). The limit of dose to the public from all anthropogenic radiation sources (excluding medical) is set at 5 mSv per year. 500 mRem – the NRC public dose is 100 m Rem/yr

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

From Public Dose Limit to RHDLs: The EPA recognizes that the recommended upper bound for dose to the public from manmade sources in a single year (5 mSv) was not developed for nuclear incidents and is not appropriate for chronic exposure (E-12). While their bases for deciding on RHDLs is not made clear, they recommend a RHDL of 20 mSv in the first year and 5 mSv thereafter.

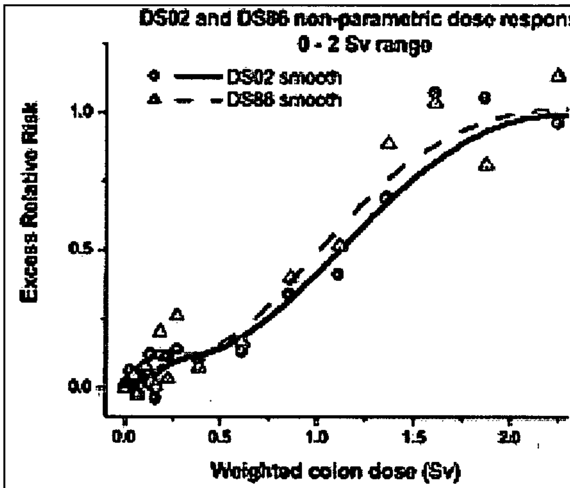
If we trace the origins of the return-home dose-limits then, we have the following steps:

- Faced with a lack of sufficient data in the relevant dose (and dose-rate) range, we use a linear extrapolation of the A-bomb data down to low doses and generate a risk of cancer fatality per Sievert.
- We determine the radiation dose needed to generate a risk of death that matches fatal risks to workers in safe industries such as trade, retail, and government; this dose becomes the maximum permissible dose to workers.
- We then divide this dose by 10 to serve as the dose limit to members of the public.
- We then increase this by 4 (first year RHDL for the EPA) or not at all (subsequent years, EPA, or first and every year, PA) in response to an unexpected or emergency-related release of radionuclides in the environment.

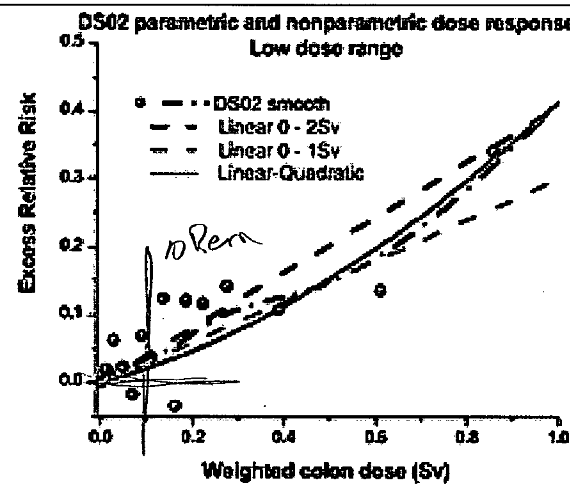
Thus, the public is not permitted to return home until the risk they are exposed to from the residual radiation is similar to 2/5 of the risk a government worker faces from an on-the-job fatality.

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

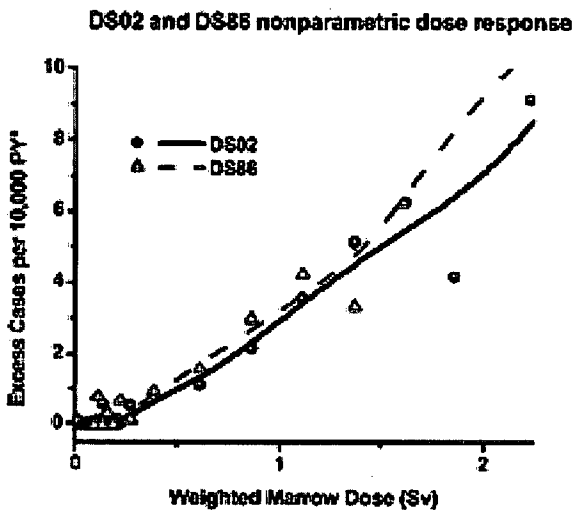
This risk, of course, is determined using simple arithmetic manipulations of the risk-per-dose estimate obtained using the LNT model as described above. Therefore, all of the occupational, public, and return-home dose limits depend very much on our assumptions about what is happening at the low dose end of Figure 2.



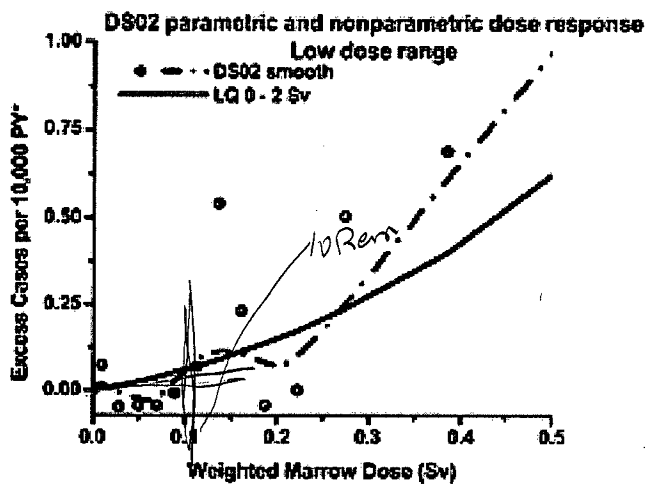
(a) Excess relative risk of solid cancer mortality.



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



(c) Excess absolute risk of leukemia mortality.



(d) Low dose range of (c) plot

Figure 2. Excess cancer risk versus dose. Plots on the left show data points using the prior dosimetry system DS86 (triangle) and the new (DS02) dosimetry calculations (circles) along with smoothed curves (b,d). An enlargement of the lower dose data is shown on the right side. Shown are the DS02 data points along with linear fits based on the dose ranges 0 – 1 Sv and 0-2 Sv and a linear quadratic fit based on the 0-2 Sv dose range.

From D.L. Preston et al, "*Effect of Recent Changes in Atomic Bomb Survivor Dosimetry on Cancer Mortality Risk Estimates*," **Radiation Research** (162) 377-89, 2004.

A.2 Uncertainty in the risk estimates:

The low dose plots of excess cancer risk estimates generated by the LSS study (shown in Figs. 2b and 2d for solid cancers and leukemia) are reproduced in Figure 3 on which the RHDLs triggered by evacuations following severe accidents at the Peach Bottom and Surry plants are superimposed. From their position on these plots it is clear that risk estimates corresponding to the RHDLs must be associated with very large uncertainties.

Uncertainty analysis of the risk-per-unit-dose estimate generated using the LNT model considers: epidemiological uncertainties (sample size, disease reporting), dosimetry uncertainty, uncertainty in transferring risk from Japanese to US populations, and projection to a lifetime of risk (since ~45% of the A-bomb survivors are still alive). Relative uncertainty estimates are generally estimated to be in the range 2 to 3 when constraining the risk versus dose relationship to pass through (0,0).

However, the LSS cancer fatality data can also be fit with models that do not pass through the origin and instead incorporate a dose threshold below which dose no excess cancer deaths are expected. In the latest evaluation of the LSS data, a threshold dose of 0.04 Sv (4 mSv) was found to fit the data with a statistical significance level *equal* to that observed when using the LNT model[6]. This is an important finding and one very relevant to the SOARCA study since the estimated health consequences of a severe reactor accident are predicted based on assuming a particular shape of the risk versus dose relationship.

NO! 10 mRem = 0.1 mSv

The potential existence of a dose threshold below which no excess cancer fatalities will occur has been considered as part of the SOARCA study. Latent cancer fatality risk was determined using (i) the LNT model, (ii) a threshold of 6.2 mSv (corresponding to the average annual radiation dose to a resident of the US from all sources), (iii) a threshold of 10 mSv (ICRP 2004), and, (iv) a threshold of 100 mSv, consistent with the recommendation of the Health Physics Society.

The use of the threshold models leads to a substantial reduction in the estimated risk of latent cancer fatalities resulting from exposure to radionuclides in the environment. However, regardless of what model is applied, it is very possible that the real consequence of a reactor accident will be the societal costs associated with the dose-avoidance strategies instigated by the relocation triggers and the RHDLs.

That is, the important point is not how we predict the number of LCFs (which we'd unlikely be able to verify anyway []) but how we approach the trade-offs between increased exposure to radiation and the recovery/mitigation actions. To address the issue of these trade-offs, we need to understand the magnitude of the hazard presented by low dose, and in particular, low dose-rate radiation.

trade off between dose and cost would need not only understanding the risk of the dose but also a display of the cost!

4 Rem

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

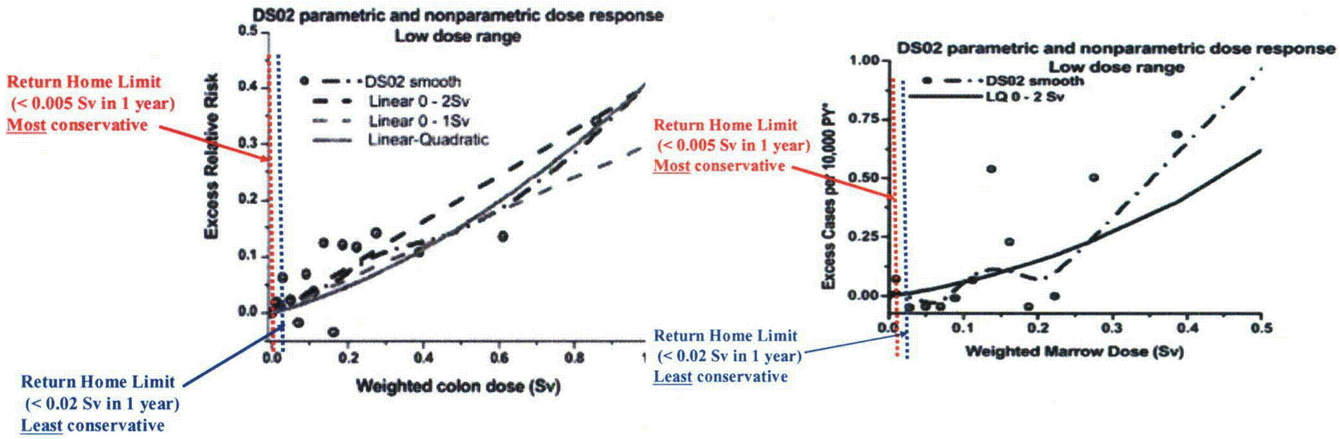


Figure 3. The RHDLs are superimposed on the low-dose cancer risk plots from Preston et al that were shown in Figs. 2b and 2d. The vertical lines representing the Pennsylvania (5 mSv) and the EPA RHDL (20 mSv) are the doses accumulated over the first year of return-home habitation. [This is an important distinction from the A-bomb doses referred to on the plot's abscissa which were delivered in less than a minute. See more on this point below.]

The importance of time: While the uncertainty associated with the shape of the dose-response curve at small doses of radiation, acutely delivered, is very large, an even greater uncertainty arises when attempting to extrapolate from the risk estimates generated using the A-bomb survivor population (for whom the entire dose was delivered in under a minute) to doses that are delivered continually, at a slow but steady rate, potentially for a lifetime.

Different biological processes proceed over a huge range of different time scales; for each process, time is an important parameter. For instance, while protein synthesis takes only seconds, repair of DNA damage needs between 30 and 120 minutes to complete. At average background dose levels in the US, each cell in our body is traversed by a secondary electron from natural radiation approximately once each year[7]. Around the world, however, there is considerable variation in soil and rock composition leading to dose-rates ranging to more than 100 times the US average. So, the cells of many people around the world experience many more hits per year, as many as 2 hits per week.

The doses to the A-bomb survivors were received in less than 1 minute. So, for those in the LSS receiving, say, a 2 Sv dose, each cell on average would have been hit ~800 times, all within 60 seconds. These variations in dose-rate are illustrated on Figure 4 which shows a logarithmic scale of dose-rate in units of both cSv/min and number of hits/cell/year. Superimposed on Figure 4(b) are the return-home dose-rates for Pennsylvania (most conservative) and Virginia (least conservative).

[7] **5 mGy = 1-2 tracks per nucleus Feinendegan numbers?**

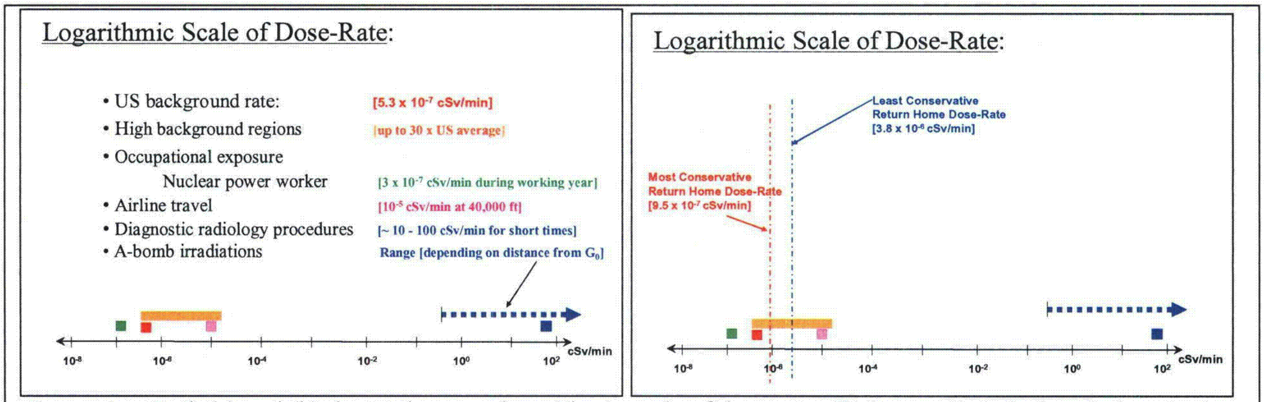


Figure 4. Both (a) and (b) depict the same logarithmic scale of dose-rate. Units are cSv/min (rem/minute). An additional scale is provided on (b) and shows number of radiation tracks (hits) per cell in the body per year (assuming 1 hit/cell/year from a whole body dose of 2.4 mSv). Shown in (a) are dose-rates from natural background, occupational dose-rates to nuclear power plant workers, dose-rates from airline travel, from diagnostic radiology, and those to the survivors of the A-bomb attacks in 1945. In (b) the dose-rates corresponding to the return-home dose-rates are superimposed on the data provided in (a).

A.3 The Dose Rate Effect:

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

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

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

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

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

Published data from laboratory studies examining dose-rate effects were examined. [Studies involving protracted (longer term) exposures were evaluated separately but it is from the shorter-term exposures that the DREF is obtained.] The dose-rate effectiveness factor (DREF) was estimated by fitting high dose-rate and low dose-rate data to linear relationships, both of which were constrained to pass through (0,0), and then taking the ratio of the slopes of these lines.

Figure 5 reproduces the logarithmic dose-rate plot shown in Figure 4(b) and superimposes the dose-rates compared in each of the studies referenced by NCRP 64 in their summary table from which the DREF for tumorigenesis is obtained.

The conclusion reached in NCRP 64 is that DREF values for tumorigenesis should lie between 2 and 10. That is, low dose-rate radiation is 2 to 10 times less effective in cancer induction than the same dose delivered acutely.

There are several points to make here:

(i) First, for the situation of long term exposure to radiation, such as the situation facing residents who return to their homes and are irradiated by the return-home dose levels, the effect of radiation protraction should be applied. This is because these residents will likely be irradiated for the remainder of their lives (albeit to an ever decreasing extent) from residual ^{137}Cs in the environment. NCRP 64 notes that:

> "protraction tends to reduce the effectiveness of the radiation exposure to a greater extent than does reduction of the dose rate."

In other words, very long-term radiation reduces the biological consequences to a greater extent than is predicted when just using the DREF.

(ii) Second, note from Fig. 5 that the lowest dose-rate examined in NCRP 64 is still a factor of 100 greater than the least conservative RHDL; most of the "low" dose-rates studied are more than 1000 times greater than the RHDLs. One of the reasons for this is the fact that biological effects at lower dose-rates could not be observed.

In fact, low doses and low dose-rates lead to *increased* longevity rather than the decreased lifespan seen at higher doses and dose-rates. In addressing the apparent life lengthening at low dose-rates, the NCRP interprets this effect as reflecting "a favorable response to low grade injury leading to some degree of systemic stimulation." They go on to state that "...there appears to be little doubt that mean life span in some animal populations exposed to low level radiation throughout their lifetimes is longer than that of the unirradiated control population. (p. 104)" hormesis

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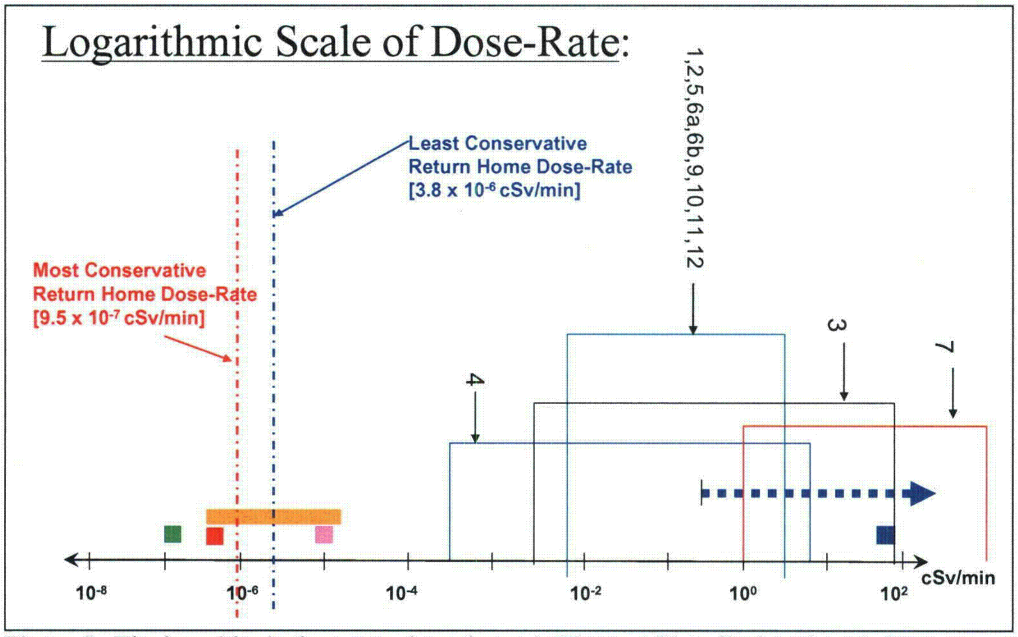

Figure 5. The logarithmic dose-rate chart shown in Figure 4(b) is displayed again here; superimposed are the dose-rate comparisons made by studies referenced by NCRP 64 in estimating dose-rate effectiveness factors for tumorigenesis. A total of 12 studies are cited; the dose-rates compared by each study are indicated, numbered here according to their number in Table 9.3, NCRP 64. Note that none of the comparisons examined dose rates similar to the RHDLs, or even within a factor of 100.

(iii) And third, the conclusions of NCRP 64 have appeared in all agency and committee discussions of the effect of dose-rate since that time. Subsequent examination of the dose-rate effect in later NCRP publications and by other committees (eg. BEIR, ICRP, UNSCEAR) rely heavily on NCRP 64 and usually invoke a DREF of 2 (or 1.5) when converting risk estimates based on acute radiation delivery to the low dose-rate scenarios of occupational exposure. The larger values quoted in NCRP 64 (ie. DREF of 2 – 10) are not typically used, presumably so as to err on the side of additional caution when it comes to radiation protection. [A DREF of 2 is used by the MACCS2 code and is therefore incorporated into the SOARCA estimates of latent cancer fatalities.] Each committee also reviews relevant studies of the dose-rate effect published since NCRP 64. Again, however due to the difficulties in actually observing deleterious effects at very low dose-rate, none of the laboratory studies examined address the dose-rates encountered upon returning home following a severe reactor accident.

no input to the MACCS2 code

To summarize:

- Our best dataset (A-bomb survivors) provides little information regarding the health effect of low doses of radiation.
- The estimated uncertainty in the risk estimates (a factor of 2 – 3) is determined using the entire dose range and by assuming the risk-versus-dose relationship passes through (0,0).
- A truncated response function with a threshold dose of 40 mSv (4 Rem) fits the LSS data with a statistical significance level equivalent to that of a linear no–threshold model. The uncertainty at the low dose region is thus much greater than a factor of 2-3.
- Extrapolating results of the LSS to situations involving long-term, low dose-rate radiation exposure introduces additional very large uncertainties.

- The DREF of 2.0 used routinely by agencies/committees developing radiation protection strategies should be increased when estimating effects from protracted exposure such as those encountered upon returning home following a severe reactor accident.
- For instance, in reviewing NCRP 64, BEIR V states that the highest values of a dose-rate effectiveness factor came from studies of the effects of continuous daily irradiation until death, but found that this "may be an unlikely circumstance for humans except as a result of natural background radiation."
- No data used in the determination of dose-rate effectiveness factors were obtained at the dose rates we can expect at the RHDLs and protraction effects are rarely examined in the risk estimates literature. Therefore we really do not know how to estimate the health consequences of these dose-rates.

As described earlier, the lack of data on which to base estimates of risk from low dose-rate radiation has not presented a major hurdle for radiation protection scenarios where the response by industries has been to simply keep the doses very low. But once an accident has occurred it is the public who is faced with both the benefits and the costs of all risk avoidance strategies. Therefore a *realistic* estimate of the hazards of low dose-rate radiation is critical so that important societal decisions can be made.

Appendix B: Recommendations

The discussion in Appendix A highlighted the enormous uncertainty we face when trying to predict the impact of chronic, low dose-rate radiation. The SOARCA study demonstrates, however, that it is these dose conditions, almost exclusively, that we will be faced with should a severe reactor accident result in elevated radionuclide levels in the environment. It is here, therefore that the NRC and the nuclear industry should devote extensive efforts.

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

We don't have the money or the reason to do this!

It cannot be in the best interests of the public to simply mandate extensive dose avoidance strategies, when the harm of not avoiding the elevated radiation is so unknown and when the costs for these avoidance strategies are significant and are borne by the public. While it is not the role of the NRC to dictate how the RHDLs should be set, the NRC and the industry should take a stronger position on determining the true nature of radiation-related health effects at the dose-rates anticipated following a severe nuclear power plant accident.

DOE is and should be, along with EPA.

If, in the event of an accident, we are to make hard decisions involving trading increased exposure to radiation against abandoning homes, public buildings, major industries, destroying livestock and crops, then we need some solid data on which to base these decisions. These data do not exist, and, further, to generate these data will require a wait-time of decades; radiation-induced solid cancers do not appear for 20-30 years following an acute exposure. Therefore, in order to generate data we need to begin now. I have three recommendations which could move us in the direction of obtaining the needed data.

1. **Systematic study of the health effects of residents of HBRAs around the world:**
There are many regions of the world whose inhabitants are already living with the dose rates represented by the RHDLs, and have been for many generations. Residents of high background radiation areas (HBRAs) do not appear to suffer adverse effects from these dose-rates (and in some cases appear to be healthier and living longer than those living in nearby control areas with lower radiation levels). However such epidemiological comparisons suffer from small sample size, incomplete dosimetry and a lack of uniformity between studies that prohibits combining of the data.

and perhaps difficulty in defining a peer group

also the fact that perhaps the dose-resistive part of the population have been weeded out over generations!

A concerted approach involving co-ordination of efforts across several countries and involving cohort or case-control studies could be undertaken. Use of common study protocols and dosimetry methods will improve the robustness of the data and allow data pooling to increase statistical power. In such a study the existing capabilities of the Radiation Effects Research Foundation (RERF) and their history of careful, long-term follow-up of the Abomb survivors would be extremely valuable.

2. **Repeat the Expert Solicitation study (NUREG/CR 6555) asking different questions:**

but we didn't use it in SOARCA.

how will this help compared with showing 4 dose-response models? If the expert elicitation mirrors the international range of opinion, this elicitation might give a way of assigning

As mentioned in the SOARCA documentation, little guidance as to how to estimate the likely health impact of low dose, low-dose-rate radiation exposure is provided by the national and international committees who examine available data. There is a large and important gap in our understanding and this gap is precisely in the area where the NRC and the nuclear industry has the greatest need for information.

The risk estimates from NUREG 6555 are used, after modification by a DREF of 2.0, in the quantification of latent cancer fatality risk in the SOARCA study. Questions in NUREG-6555 (put to 13 experts) deal with cancer incidence and cancer death rates for a given radiation dose. For all but one case involving exposure to low LET radiation, the experts were asked to provide their estimate of the risk of fatal cancer resulting from high dose, high dose-rate radiation (1 Gy delivered over 1 minute).

DDB — The range should still cover what we show now.

However, of much greater use would be expert estimates of the effects of low dose-rate and low dose gamma radiation, ie. conditions identical to those we will encounter upon returning home following a reactor accident. [An abundance of recent data relevant to these questions comes from: epidemiological studies of people living in high background radiation areas, studies of life-long irradiation of small animals, and radiobiological examination of low dose and low dose-rate radiation on cells and tissues in the laboratory. The field of radiobiology has undergone rapid changes in the last decade with advancements in biological assays and interrogation methods that make it possible to begin addressing biological responses at lower doses than possible in decades past.]

but #1 would this improved because it's so big!

A study similar to that reported in NUREG 6555 should be performed. Expert opinion should be solicited regarding the risks associated with long-term, protracted radiation exposures of the public. Such risk estimates will be far more useful in generating a 'state-of-the-art' estimate of the consequences of elevated radionuclides in the environment.

3. Establish a Radiation Registry for EVERYONE

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

A couple of decades ago the doses received from diagnostic radiology were relatively small and experienced by relatively few individuals. Today, however, radiological exams are used for addressing a much broader range of medical questions and are performed on a much bigger fraction of the population. More important is the fact that we've begun to make routine use of the more dose-intensive procedures of x-ray computed tomography (CT) and interventional fluoroscopy. The result is that the average US resident receives as much radiation dose from diagnostic radiology procedures as from all natural background radiation sources, combined.

Thus, on a routine basis, and for a variety of reasons, we deliberately and carefully irradiate most members of the U.S. population, exposing them to a wide range of doses depending on the reason for the exam, the part of the body being imaged, and the patient's body thickness. It makes sense to maintain a registry of radiation doses for everyone irradiated. This registry would not be a "de-identified" patient radiation dose data-base (as proposed recently by the FDA as a starting point for establishing consistent exam parameters across medical institutions), but a registry that allows tracking of an individual's dose over time and, ultimately, for correlation of dose with disease or health status many years later.

Initiating and maintaining a database of patient doses, if done correctly, would provide the single greatest database for low dose radiation exposures. And while we would not be learning about the effects of low dose-rate radiation (doses from diagnostic radiology are low but the rate of radiation delivery is similar to that experienced by the A-bomb survivors], such a registry would present important advantages for risk determination not available with the LSS study. First, it is unlikely the A-bomb survivor data will ever be able to provide a statistically significant determination of the risk of radiation-induced cancer as a function of dose in the low dose range. Of course, stratification of risk estimates to ask questions about radiosensitivities based on age, gender, medical status, radiation history, etc., will also never be possible in the low dose range. On the other hand, hundreds of millions of radiological exams are performed each year in the U.S. on individuals of all ages. While some radiological procedures are performed on patients with an underlying medical condition that could represent a confounding factor for any future analysis of radiation-induced health consequences, many procedures are performed with "no evidence of disease". Even if only a subset of the radiological exams were appropriate for long-term evaluation of the effects of radiation on health we will still quickly accumulate a sufficient sample size for the statistical power we need to answer the question: what impact do low levels of radiation have on our health? With over 350 million diagnostic radiology or nuclear medicine exams performed in the U.S. in 2006 alone the statistical precision possible is very quickly greater than that with the Abomb survivor study, and we will be able to stratify the data so that we can assess the impact of low dose radiation on potentially sensitive subgroups within the population.

Is there a range of radio-sensitivities apart from these issues (that is, given for all of these individuals) is there an inherent difference in radiosensitivity?

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

? antecedent? malnourished + weakened?

There are significant hurdles to overcome in establishing such a database; however I believe it represents our best opportunity for developing an understanding of low dose effects and for this reason a strong effort should be made to overcome these hurdles. Of great importance to this effort is the current move toward digitization of patient records. This is therefore an opportune time for determining the precise parameters to capture for inclusion in the database. And, while we do not currently record the doses received by individual patients, strategies do exist for determining individual organ doses for each patient from each procedure[8] and such information could be stored in the patient's electronic record. Experience with life-long tracking of radiation dose is available with radiation registries used to track occupational radiation doses, and of course, extensive high quality analysis of long-term health effects from radiation exposure as a function of age and gender has been performed for decades by the Radiation Effects Research Foundation in their analysis of the participants in the Life Span Study. This existing experience would represent a valuable starting point for a registry of individual patient doses.

Not only 'radiation' would be needed!

would also need to take account of life-style issues (eg smoking, alcohol, drugs?) for Research registry

Appendix C;

Glossary of Acronyms:

RHDL – Return home dose limits
NAS BEIR – National Academies of Sciences Biological Effects of Ionizing Radiation
NCRP - National Commission on Radiological Protection
ICRP – International Commission on Radiological Protection
EPA – Environmental Protection Agency
UNSCEAR – United Nations Scientific Committee on the Effects of Atomic radiation
DS02 – Dosimetry Study 2002
ALARA – As Low as Reasonably Achievable
LSS – Life Span Study
LET – Linear Energy Transfer
HBRA – High Background Radiation Area
STSBO – Short term station black out

might also need to keep track of the diagnosis from the procedure, so that confounding features of any treatment can be taken into account.

Units of Dose (and effective dose equivalent):

500 mrem	= .005 Sv = 5 mSv
2 rem	= 0.02 Sv
100 rad	= 1 Gy
100 rem	= 1 Sv
1 Gy	= 1 Sv for the low LET radiation arising from gamma emitters in the environment, as discussed here.

[8] Currently we do not record the doses received by individual patients. Instead what we do is take note of how many radiological exams of a particular type are performed each year in the country, then, on a one-time basis, we measure the dose to a 'typical' patient (using a human "Mock-up") from this exam. We then multiply the two values together for an estimate of the dose to the entire population, on average, from this particular exam.

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).	will have to address
7/29/09	Bixler pres., slide 9	Yanch	Explain why the RBE for bone marrow is reduced to 1.	will have to address
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".	would require other changes - e.g. separation in treatment of children -
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	discuss that they are the same - pick one + fix others
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. done	

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.	check & make sure NUREG discussion is now adequate
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.	NO!
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?	I believe they are the same check + make sure NUREG explains
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.	It will be evaluated in the uncertainty [illegible] along with economic impact!)
7/29/09	Bixler pres., slide 33	Gabor	Highlight qualitatively the differences between SOARCA and SST1 results and the general reasons for the differences.	NO! should take it out altogether since it is misleading
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?	not as LNT [illegible]!

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?

See page 7 where Vierow believes that MELCOR has been benchmarked against TMI-2

need to address

[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.

But there wasn't a paper exercise for PB since equipment/training etc not available

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.

yes!

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. NO!

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.

yes, but a what cost?

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.	what was this? why omit "negative perspective" data Need to explain in the text -
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.	yes, but then need to discuss other (cost) implications
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.	with 2 metrics, no single weather is bounding!
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.	how can 2 specific scenarios be used to justify the mean of ~1000?
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.	need to ensure a complete discussion
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)}.	No reduction in latent risk!

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.

but they would qualify than a sequence below the cutoff that [illegible] consequence significantly exceeding those above the cutoff.

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.

Have pursued the Ba? Check what was result.

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? probably, but don't know if we edited it
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: Don't understand
    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. smaller numerical value?
    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. 100 mrem PA or 400 mrem Surry?
    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? ad-hoc evac is only for >10 miles
    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.

will do for uncertainty study

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.

good question

j. What is the effect of degradation of containment leaktightness due to an earthquake in the 0.3 to 0.5 pga range, and in the 0.5 to 1.0 pga range? For example, consider DBA leakage x3 for 0.3 to 0.5 pga, and x10 for 0.5 to 1.0 pga.

k. This matter was brought up in one of the first two meetings by Jeff Gabor. What about a sensitivity on the radionuclide release assuming that the SRV sticks closed after excessive cycles (see Volume III, Figure 31)?

October 6, 2009

(From John Stevenson)

Dear Karen:

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

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

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

- Seismic Issue

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

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

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

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

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

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

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

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

- Hydrogen

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

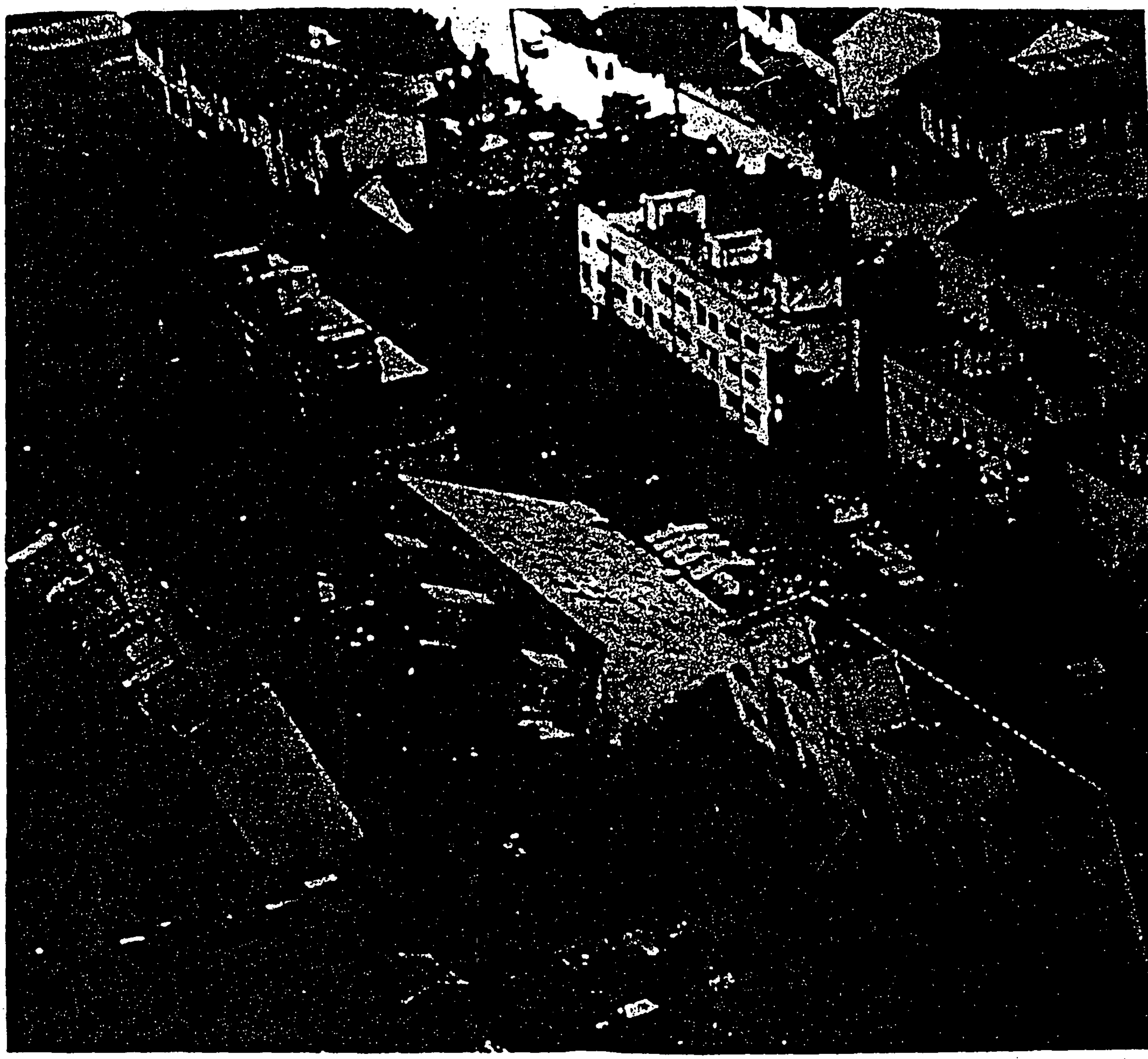

FIGURE 9 TILTING OF APARTMENT BUILDINGS AT KAWAGISHI-CHO DUE TO SOIL LIQUEFACTION RESULTING FROM THE NIIGATA EARTHQUAKE, JUNE 16, 1964.

Comments/Questions on SOARCA Volume I

COMMENT AND RESOLUTION SHEET			
Document Title: State-of-the-Art Reactor Consequence Analysis (SOARCA) Project SOARCA Methods Volume I	**Doc. No. NUREG-XXXX/SAND2008P-XXXX**	**Doc. Date:** July 2009	**Date Comment Sent:** 21 August 2009
Commenting Individual or Organization: Kevin O'Kula, WSMS	**Phone**/Email: 803.502.9620/kevin.okula@wsms.com	**Resolution by:**	**Phone No.**

No.	Page	Section	Type: M – Major Med - Medium Min –Minor E -- Editorial	Comment, Question	Resolution of Comment
1	ix	Background and Objective	Med.	NUREG/CR – 2239 and NUREG/CR- 2723 are both cited as being referenced throughout the SOARCA documentation. I have been able to download the latter (Strip report) but the former (Sandia Siting Study) does not appear to be available on the web. Can this be made available to the review panel?	
2	xi		Editorial	2[nd] paragraph, 2[nd] line: American Society of Mechanical Engineers'	
3	3	1.0 Introduction	Editorial	A introductory, transition sentence or two is needed ahead of the first paragraph on page 3. The paragraph reads as though it is the present tense, e.g. " Yet the possibility remains . . .". Suggest a statement to note that it is in reference to the state of knowledge during or after WASH-1250.	
4	15	Section 2.2	Minor	Suggest that first use of SPAR models be noted with a citation/reference.	
5.	22	3.11	Minor	Was short-term Station Blackout from a seismic event for Peach Bottom included or dropped?	given the uncertainty in other parameters we are already doing a dependency fit on met. why do more?
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)?	how would you do LHS? It would require producing a CDF, discretizing into equi-likely bands, and MC sampling within

w/ 10K 4 parameters, how do you produce a CDF? would require coding.

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**
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 VSP results 3X the LCF	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.	It will be in the uncertainty study
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?**	Should be in the uncertainty study - check to add
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? no	separately!
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.**	Should add	

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.	Too open ended a request; since we changed DCFs and radionuclides, hundreds of changes were made.

Comments/Questions on SOARCA Volume III

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

No.	**Page**	**Section**	**Type: M – Major Med - Medium Min –Minor E -- Editorial**	**Comment, Question**	**Resolution of Comment**

Comments/Questions on SOARCA Volume IV

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

Comments/Questions on SOARCA Volume IV

COMMENT AND RESOLUTION SHEET					
Document Title: State-of-the-Art Reactor Consequence Analysis (SOARCA) Project SOARCA Surry Integrated Analyses Report Volume IV				**Doc. No. NUREG-XXXX/SAND2008P-XXXX**	**Doc. Date:** July 2009 / **Date Comment Sent:** 20 August 2009
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**

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. curb

   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.

maybe it does, it makes it more likely that the LOCA piping will not fail

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

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

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

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

   NO, only bypass frequency - not any early containment failure

   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.
   yes, we had to do
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.

good point - AJ has been raising this issue for some time

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
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(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?

<u>*Jacquelyn Yanch and David Leaver have provided the following comments.*</u>

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.

source?

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Kevin O'Kula added the following clarification.

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

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

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

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

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

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

There is the possibility that certain accident sequences, while not-dominant from a frequency basis, may have relative high risk due to high consequence. While these sequences may not dominant the risk, in terms of frequency and consequence, they could be contributors. Collections of several lower order sequences, while individually non-dominate, could have higher consequence than SOARCA evaluated and could contribute to the risk collectively. While SOARCA did indeed capture the most likely sequences and accurately capture the consequence from these sequences, the primary issue with consequence analyses of this type is that it is difficult to *demonstrate completeness. A Level 3 PRA performed for the SOARCA plant could have the benefit of reduced resources (due to work performed for SOARCA) as well as the benefits of validation of the SOARCA approach and demonstration of completeness. For these reasons, a Level 3 PRA for the SORCA plant might have some value.*

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

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

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

John Stevenson recommends an evaluation of plant foundation conditions.

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

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

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

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

Bruce Mrowca has not provided an opinion.

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

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

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

Memorandum:

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

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

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

Date: 9 March 2010

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

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

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

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

document value

International Commission on Radiological Protection (1999) Risk Estimation for multifactorial diseases. Ann. ICRP 29:1-144.

United Nations Scientific Committee on Effects of Atomic Radiation (2000) Sources and Effects of Ionizing Radiation. UNSCEAR Report to the General Assembly.

Environmental Protection Agency (1994) Estimating Radiogenic Cancer Risks, EPA Report 402-R-93-076. Washington DC: Environmental Protection Agency.

National Research Council (1990). Health Effects of Exposure to Low Levels of Ionizing Radiation (BEIR V). Washington DC: National Academy Press.

National Research Council (2006). Health Effects of Exposure to Low Levels of Ionizing Radiation (BEIR VII). Washington DC: National Academy Press.

EPA (1999) Cancer Risk Coefficients for Environmental Exposure to Radionuclides. Federal Guidance Report No.13.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Appendix E Memo on Uncertainty Quantification and Sensitivity Analysis

MEMO

Re: Guidance on the SOARCA Uncertainty Quantification and Sensitivity Analysis

To: SOARCA Team

Through: S. P. Burns

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

Date: April 9, 2010

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

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

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

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

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

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

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

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

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

Bruce Mrowca has not provided an opinion.

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

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

NUREG – 1935
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State-of-the-Art Reactor Consequence Analyses (SOARCA) Project

Summary Report

Manuscript Completed: XXXX
Date Published: XXXX

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555-0001

Sandia is a multi-program laboratory operated by Sandia Corporation, a Lockheed Martin Company, for the United States Department of Energy's National Nuclear Security Administration under contract DE-AC04-94AL85000.

ABSTRACT

or leave that in and add; As part of the SOARCA project, excessively non-conservative aspects of the previous analyses were also addressed,

The evaluation of accident phenomena and the offsite consequences of severe reactor accidents has been the subject of considerable research by the NRC over the last several decades. As a consequence of this research focus, analyses of severe accidents at nuclear power reactors is more detailed, integrated and realistic than at any time in the past. A desire to leverage this capability to address ~~excessively conservative~~ aspects of previous reactor accident analysis efforts was a major motivating factor in the genesis of the State-of-the-Art Reactor Consequence Analysis (SOARCA) project. By applying modern analysis tools and techniques, the SOARCA project seeks to provide a body of knowledge that will support an informed public understanding of the likely outcomes of severe nuclear reactor accidents. 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 utilized integrated modeling of accident progression and off site consequences using both state-of-the-art computational analysis tools as well as best modeling practices drawn from the collective wisdom of the severe accident analysis community.

selected

unrealistic

Paperwork Reduction Act Statement

The information collections contained in this NUREG are covered by the requirements of 10 CFR Parts 50, 52, and 110, which were approved by the Office of Management and Budget, approval number 3150-0011, -0151 and -0036.

Public Protection Notification

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ACKNOWLEDGEMENTS

The contributions of the following individuals in preparing this document is gratefully acknowledged.

Jon Ake	Nuclear Regulatory Commission
Nathan E. Bixler	Sandia National Laboratories
Jeffrey D. Brewer	Sandia National Laboratories
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Kenneth C. Wagner	Sandia National Laboratories

Scott Elkins?
Robert Prato?

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TABLE OF CONTENTS

FIGURES

TABLES

ACRONYMS

AC	Alternating Current
AFW	Auxiliary Feedwater
ATWS	Anticipated Transient Without Scram
CCI	Core Concrete Interactions
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CST	Condensate Storage Tank
DC	Direct Current
DHS	Department of Homeland Security
EAL	Emergency Action Levels
ECCS	Emergency Core Cooling System
ECST	Emergency Condensate Storage Tank
EPZ	Emergency Planning Zone
ETE	Evacuation Time Estimate
FR	Federal Register
GE	General Emergency
HPCI	High Pressure Coolant Injection
HPI	High Pressure Injection
IPEEE	Individual Plant Examination – External Events
ISLOCA	Interfacing Systems Loss-of-Coolant Accident
LLNL	Los Alamos National Lab
LOCA	Loss Of Cooling Accident
LOOP	Loss Of Offsite Power
LPI	Low Pressure Injection
LTSBO	Long Term Station Blackout
MSIV	Main Steam Isolation Valve
NG	Noble Gas
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
OREMS	Oak Ridge Evacuation Modeling System
ORO	Offsite Response Organizations
PGA	Peak Ground Acceleration
PRA	Probabilistic Risk Assessment
PORV	Power operated relief valve
RCIC	Reactor Core Isolation Cooling
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RWST	Refueling Water Storage Tank
SAE	Site Area Emergency
SBO	Station Blackout
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SOARCA	State-of-the-Art Reactor Consequence Analysis Project

BEF BEIR
CPI
BP DCF
BOP
EOP
FBMA
IPE
LCF LET
LPCS LNT
NRF
RHR
SAMG

SORV	Stuck Open Relief Valve
SPAR	Simplified Plant Analysis Risk
SRM	Staff Requirements Memorandum
SRV	Safety Relief Valve
STSBO	Short Term Station Blackout
TAF	Top of Active Fuel
TD-AFW	Turbine Driven Auxiliary Feedwater
UE	Unusual Event

STCP
TSC
WHD

1.0 INTRODUCTION

This document describes the U.S. Nuclear Regulatory Commission's (NRC's) state-of-the-art, realistic assessment of the accident progression, radiological releases and offsite consequences for important severe accident sequences. The primary focus of this volume is to provide the background and objectives of the study and summarize~~s~~ the methods used to perform the analysis. Appendices A and B discuss the detailed modeling practices and the plant-specific results. and selected results of those analyses all of

1.1 Purpose of SOARCA

The evaluation of accident phenomena and offsite consequences of severe reactor accidents has been the subject of considerable research by the NRC, the nuclear power industry and the international nuclear energy research community. Most recently, with Commission guidance and as part of plant security assessments, updated analyses of severe accident progression and offsite consequences were completed utilizing the wealth of accumulated research, which are more detailed – in terms of the fidelity of the representation and resolution of facilities and emergency response, realistic – in terms of the use of currently accepted phenomenological models and procedures, and integrated – in terms of the intimate coupling between accident progression and off-site consequence models. The results of these recent studies have confirmed and quantified what was suspected but not well-quantified — namely, that some past studies of plant response and offsite consequences were conservative to the point that predictions were not useful for characterizing results, communicating to the public or guiding public policy. The subsequent misuse and misinterpretation of these estimates further suggests that communication of risk attributable to severe reactor accidents should be based on realistic estimates of the more likely outcomes. Further, as a result of past risk assessments and in response to the terrorist attacks of September 11, 2001 nuclear plants have made additional safety enhancements which reduce the risk of severe accidents as portrayed in earlier NRC assessments.

the source term and fatality aspects of

The objective of the State-of-the-Art Reactor Consequence Analysis (SOARCA) project is to develop the updated estimates of potential site-specific offsite consequences from severe accidents for operating nuclear power plants (NPPs). The Peach Bottom Atomic Power Station and the Surry Power Station were the first two plants selected to perform ~~risk-informed~~ consequence analyses. The licensee provided detailed information on the current plant designs and configurations including their existing and newly developed mitigative measures to the extent practical. The analyses were performed using state-of-the-art thermal-hydraulic severe accident progression modeling, state-of-the-art consequence analysis modeling, and the latest emergency preparedness assumptions and criteria to assess the potential effects to public health and safety in the unlikely event of a severe accident at an operating U.S. nuclear power plant. The analyses were used to determine the average probability of an individual dying from acute exposure or latent cancer conditional on the occurrence of a severe reactor accident.

core damage frequency informed

the most likely

The results from the SOARCA project to date provide an updated reference of the likely outcomes of severe reactor accidents at the Peach Bottom and Surry sites, based on the most current emergency preparedness (EP) and plant capabilities. It is also anticipated that the study will be a resource for future modeling improvements and verification efforts.

and core degradation

1.2 Background

than

In the first decade of nuclear power, the reactors were low power and of experimental designs. The fission product inventories and heat removal requirements of these low power reactors were much lower of those today. As newer designs approaching 500 MW_{th} were developed, the Atomic Energy Commission (AEC) began serious studies of accidents and their consequences. Over the following 40 years the AEC and later the NRC would produce a number of reports that examined the broad spectrum of reactor risk and consequence. Each study built upon the prior study and added newer research and experience to sharpen the models of nuclear accidents.

In addition to the improvements in understanding and calculational capabilities that have resulted from these studies, 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 for the boiling water and pressurized water reactor designs;
- the performance and maintenance of plant-specific probabilistic risk assessments 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; and
- proceduralized use of plant specific systems required under Title 10, Section 50.54(hh) of the Code of Federal Regulations; ~~and~~
- ~~improved phenomenological understanding of influential processes such as~~
  - ~~in-vessel steam explosions,~~
  - ~~Mark I containment liner attack,~~
  - ~~dominant chemical forms for fission products,~~
  - ~~direct containment heating,~~
  - ~~hot leg creep rupture,~~
  - ~~reactor pressure vessel failure, and~~
  - ~~molten core concrete interactions.~~

Or increase the scope given for the list of and more aspects - e.g. EP health effects etc -

This does not seem to fit the category of "training" and "utilization of plant specific capabilities"

The following sections describe the contributions of the seminal nuclear safety works sponsored by the AEC and NRC to this body of knowledge.

1.2.1 WASH-740, Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants, 1957

An important technical input to establishing the indemnity provisions of the Price-Anderson Act was the report WASH–740 [1], which was prepared by Brookhaven National Laboratory and published by the AEC. Using what would later prove to be extremely pessimistic assumptions including a core meltdown with the release of fifty percent of the core fission products to the atmosphere, the worst case consequences of a 500 MWt reactor accident were estimated to be 3,400 early fatalities, 43,000 acute injuries, and a 7 billion (1957) dollar financial impact. There was a consensus among those involved in the WASH-740 study that the likelihood of a meltdown accident was low, but quantitative probability estimates could not be supported given the lack of operating plant experience. Similarly, the likelihood of containment failure (or bypass) given a meltdown accident was not quantified (or quantifiable, at the time). However, until 1966, the containment building was treated as an independent barrier, which should remain intact even if the core melted, thereby preventing any large release of radionuclides to the atmosphere. It was recognized that failure of the containment building and melting of the core could occur—for example, as a consequence of gross rupture of the reactor pressure vessel—but such events were not considered credible. Containment failure was not expected to occur simply because the core melted.

1.2.2 WASH-1250, The Safety of Nuclear Power Reactors (Light Water-Cooled) and Related Facilities, 1973

Senator John Pastore requested a comprehensive assessment of reactor safety. The AEC's first response to this request was the WASH-1250 report [2], which was published in final form in July 1973. WASH-1250 provided factual information regarding the conservatisms applied in the design of nuclear power plants. It did not, however, address the likelihood or potential consequences of beyond-design-basis accidents, that is, failures beyond those postulated under the single failure criteria.

1.2.3 WASH-1400, (NUREG-75/014), Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, 1975

(PRA)

Several questions had to be addressed in order to respond to Senator Pastore's request for a comprehensive assessment of reactor safety. What accidents could result in significant core damage and containment breach or bypass? How likely are these accidents? What would be their health and economic consequences? These are fundamental questions that WASH-1250 did not address. Such questions are addressed in probabilistic risk assessments, but, at the time, had not to the techniques of PRA been applied to NPPs.

In the summer of 1972 the AEC initiated a major probabilistic study, the Reactor Safety Study. Professor Norman C. Rasmussen of the Massachusetts Institute of Technology served as the study director. Saul Levine of the AEC served as staff director of the AEC employees that performed the study with the aid of many contractors and consultants.

The study's team attempted to make a realistic estimate of the potential effects of light water reactor (LWR) accidents on the public health and safety. One BWR, Peach Bottom Unit 2, and one PWR, Surry Unit 1, were analyzed in detail to estimate the likelihood and consequences of potential accidents. These plants were chosen because they were the largest plants of each type that were about to start operation.

The study's stated purpose was to quantify the risks to the general public from commercial NPP operation. This logically required identification, quantification, and phenomenological analysis of a wide range of low-frequency, relatively high-consequence scenarios that had not previously been considered in much detail. ~~The introduction at this point of the concept of "scenario" is significant; as noted above,~~ Many design assessments simply look at system reliability (success probability), given a design basis challenge. The review of nuclear plant license applications did

can't find a note above

essentially this, culminating in findings that specific complements of safety systems were single-failure proof for selected design basis events. Going well beyond this, WASH-1400 [3] modeled "scenarios" leading to large radiological releases from each of the commercial NPPs considered. It considered highly complex scenarios involving success and failure of many and diverse systems within a given scenario, as well as operator actions and phenomenological events.

The team adapted methods previously used by the Department of Defense and the National Aeronautics and Space Administration to predict the effect of failures of small components in large, complex systems. The overall methodology, which is still utilized, is called probabilistic risk assessment (PRA).

The team first identified events that could potentially lead to core damage. Event trees were then used to delineate possible sequences of successes or failures of systems provided to prevent core meltdown and/or the release of radionuclides. Fault trees were used to estimate the probabilities of system failures from available data on the reliability of system components. Using these techniques, thousands of possible core melt accident sequences were assessed for their occurrence probabilities. The public health and economic consequences of the identified severe accidents were estimated using computational models that were developed as part of the overall effort.

The insights gained from the Reactor Safety Study included: (a) melting of the reactor core does not necessarily result in an accident having large public consequences, (b) the number of fatalities expected from the most likely course of events following a melting of a core is much smaller than those that commonly occur in accidents such as fires, explosions and crashes of a commercial jet airplane, and (c) there are wide varieties of weather conditions and population densities where reactors are located and when appropriate frequencies of occurrence are assigned, these can cause potential accident consequences to increase by 100 to 1000 times, however, the probability of such accidents could decrease by generally similar factors.

does not make a lot of sense. I apply frequencies and the variety of weather and pop could make consequences go up by 100-1000; but then the frequency could decrease

It was assumed that there would be 100 power reactors and that they all had risks equal to the average risks for Surry and Peach Bottom. This assumption was not rigorously investigated. In particular the study stated as a limitation that it would not be appropriate to extrapolate the results beyond 100 reactors and 5 years. This limitation was based on the observation of continued attention to improved safety. The assumed improvement depended strongly on the continuing existence of competent and well supported regulatory and reactor safety research programs and reasonably conservative extrapolation of current practice.

While the risks from nuclear power appear to be very low, the Reactor Safety Study did indicate that core melt accidents were more likely than previously thought (approximately 5×10^{-5} per reactor year for Surry and Peach Bottom), and that light water reactor risks are mainly attributable to core melt accidents. The Reactor Safety Study also demonstrated the wide variety of accident sequences (initiators and ensuing equipment failures and/or operator errors) that have the potential to cause core melt. In particular, the report indicated that, for the plants analyzed, accidents initiated by transients or small LOCAs were more likely to cause core melt than the traditional large design-basis LOCAs.

Loss of coolant accident

In addition to providing some quantitative perspective on severe accident risks, WASH- 1400 provided other results whose significance has helped to drive the increased application of PRA in the commercial nuclear power arena. ~~It showed, for example, that some of the more frequent, less-severe initiating events (IEs), e.g., "transients," lead to severe accidents at higher expected frequencies than do some of the less frequent, more severe IEs (e.g., very large pipe breaks). It~~ led to the beginning of the understanding of the level of design detail that must be considered in PRA if the scenario set is to support useful findings (e.g., consideration of support systems and environmental conditions). Following the severe core damage event at Three Mile Island (TMI) in 1979, application of these insights gained momentum within the nuclear safety community, leading eventually to a PRA-informed re-examination of the allocation of licensee and regulatory safety resources. In the 1980s, this process led to some significant adjustments to safety priorities at NPPs; in the 1990s and beyond, regulation itself is being changed to refocus attention on areas of plant safety where that attention is more risk important.

already said this bottom of last page or leave this here and take out last page -

1.2.4 NUREG/CR-2239, Technical Guidance for Siting Criteria Development, 1982

Following the TMI accident, NRC contracted with Sandia National Laboratories to develop a technical guidance report for siting future reactors [4]. Guidance was requested regarding (1) criteria for population density and distribution surrounding future sites, and (2) standoff distances of plants from offsite hazards. (SST)

Since the purpose was to develop criteria for siting by evaluating aspects of population and meteorology separately, 5 types of accidents ~~would~~ were to be imposed on each plant in the 92 ~~plant~~ site study. The accidents or "siting source term events" would be derived from the previous Reactor Safety Study (WASH-1400) and each SST event would be assumed identical regardless of the reactor size or plant design. Although the absolute numerical results may be questionable due to the arbitrary source terms, the relative impact of population density, weather, and evacuation times would be apparent for every site in the United States.

the siting source terms were defined at the study's sites where there were actual plants.

SST1 – Severe core damage. Loss of all safety systems and loss of containment after 1.5 hours.

SST2 – Severe core damage. Containment systems (e.g., sprays, suppression pools) function to reduce radioactive release but containment leakage is large after 3 hrs.

SST3 – Severe core damage. Containment systems function but small containment leakage (1 % per day) after 1 hr.

SST4 – Modest core damage. Containment systems function but small containment leakage after ½ hr.

SST5 – Limited core damage. Containment functions as designed with minimal leakage.

The results for most of the 92 ~~reactor plants~~ sites were similar due to a low population density. Using the SST1 model with a population density of 50 persons per square mile resulted in 47 to 140 early fatalities and 730 to 860 latent cancer fatalities. For the more realistic release represented by SST2 events, the mean values from typical plants were zero early fatalities and 95-140 latent cancer fatalities.

1.2.5 NUREG-1150, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, 1990

NUREG-1150 [5] documents the results of an extensive NRC-sponsored PRA. The study examined five plants, representative of classes of reactor and containment designs to give an understanding of risks for these particular plants. Selected insights regarding the classes of plants were also obtained in the study. The improved PRA methodology used in the NUREG-1150 study significantly enhanced the understanding of risk at nuclear power plants, and is considered a significantly updated and improved revision to the Reactor Safety Study. A major improvement was the specific inclusion of an uncertainty estimate for the core damage frequency and source term portions of the study, but not for the off-site consequence portion. The uncertainty estimate was based on extensive use of expert elicitation.

The five nuclear power plants analyzed in NUREG-1150 are:

- Unit 1 of the Surry Power Station, a Westinghouse-designed three-loop reactor in a sub atmospheric containment building, located near Williamsburg, Virginia;
- Unit 1 of the Zion Nuclear Power Plant, a Westinghouse-designed four-loop reactor in a large, dry containment building, located near Chicago, Illinois;
- Unit 1 of the Sequoyah Nuclear Power Plant, a Westinghouse-designed four-loop reactor in an ice condenser containment building, located near Chattanooga, Tennessee;
- Unit 2 of the Peach Bottom Atomic Power Station, a General Electric designed BWR-4 reactor in a Mark I containment building, located near Lancaster, Pennsylvania; and
- Unit 1 of the Grand Gulf Nuclear Station, a General Electric-designed BWR-6 reactor in a Mark III containment building, located near Vicksburg, Mississippi.

were The various accident sequences that contribute to the core damage frequency from internal initiators ~~can be~~ grouped by common factors into categories. NUREG-1150 use~~s~~d the accident categories depicted in Table 1 below: station blackout, anticipated transients without scram (ATWS), other transients, interfacing system LOCAs, and other LOCAs. The selection of such categories is not unique, but merely a convenient way to group the results.

Table 1 Summary of Core Damage Frequency from NUREG-1150

Plant Name	Internal Initiators						External Initiators
	SBO	ATWS	TRANS	SG/IF Sys	LOCA†	Core Damage Total/yr	Fire & Seismic
Surry	2.7E-5	1.6E-6	2.0E-6	3.4E-6	6.0E-6	4.0E-5	2.6E-5

Peach Bottom	2.2E-6	1.9E-6	1.4E-7	-	2.6E-7	4.5E-6	2.3E-5

† The LOCA category shown here includes LOCAs that are initiated by pipe break events. Transient-induced LOCAs are captured under the other categories shown in the table.

1.3 Objectives

The basic approach for the SOARCA project was to utilize the self consistent, integrated modeling of accident progression and offsite consequences drawn from current best practices modeling, to estimate offsite consequences for important classes of events. This was accomplished by modeling accident progression (reactor and containment thermal-hydraulic and fission product response), which is embodied in the MELCOR code, coupled with modeling offsite consequences (in the MACCS2 code) in a consistent manner (e.g., accident timing) and with improved input in important areas. Selection of the events for analysis was based on a consideration of insights from past and current PRA, and from research on accident behavior and failure modes important to latent cancer and early fatality risk. Selection of events for quantification also properly included probability in order to focus on more likely and important contributors. It is believed that more can be learned at this juncture by focusing on a relatively few important events and quantifying the plant and offsite response rigorously and realistically than by approximate modeling of many events, including extremely rare events. This approach of focusing on a relatively few, but important events, also allows us to efficiently and explicitly address the benefits of additional mitigation in further reducing the likelihood of core damage and offsite consequences. The off-site consequence analyses were performed on a site-specific basis (reflecting site-specific population distributions, weather, and EP) and also included improved understanding of non-site specific input. emergency preparedness

Selection of events considered individual plant examinations (IPEs), individual plant examinations of external events (IPEEEs), standardized plant analysis risk (SPAR) models, and NUREG-1150 risk studies. Information related to system and procedural plant improvements that have been incorporated as part of the industry's response to the NRC's security initiatives, (e.g., the purchase and development of procedures for diesel-driven pumps in response to 10CFR50.54(hh) requirements), as well as necessary plant information ~~was~~ were included in the scenario selection evaluation and incorporated in plant modeling.

1.4 Scope

The central focus of the SOARCA project was to introduce the use of a detailed, best-estimate, self-consistent quantification of sequences based on current scientific knowledge and plant capabilities. The essence of the analysis methodology is the application of the integrated severe accident progression modeling tool, the MELCOR code. The analysis used an improved off-site consequence (MACCS2) code, including both improved code input and updated sequence-specific emergency response. Because the priority of this work was to bring more detailed, best-estimate, and consistent analytical modeling to bear in determining realistic outcomes of severe accident scenarios, the benefits of this state-of-the-art modeling could most efficiently be demonstrated by applying these methods to a set of the more important severe accident sequences. Thus the project elected to limit its analysis to a set of important accident sequences considering both likelihood and potential consequences. The sequences which eventually were selected (e.g., station blackout, ISLOCA, thermally induced steam generator tube rupture) are in fact sequences which also have been considered to be important in recent and past probabilistic assessments.

There were several classes of accident events which were not considered as part of the SOARCA project as identified below:

- Multi-unit accidents
- Low power and shutdown accidents
- Extreme seismic events which lead directly to gross containment failure simultaneous with reactor core damage
- Spent fuel pool accidents
- Security events

Multi- unit accidents (events leading to reactor core damage at multiple units on the same site) could be caused by certain initiators such as an earthquake. Most PRA's developed to date do not explicitly consider multi-unit accidents because NRC policy is to apply the Commission's Safety Goals (51 FR 28044) [6] and subsidiary risk acceptance guidelines (see Regulatory Guide 1.174) on a "per reactor" basis [7]. Therefore, no multi-unit accident scenarios were selected for the SOARCA project. ~~The results of the unmitigated sequence analysis in SOARCA suggest that consideration of multi-unit accidents would not substantially alter the study findings regarding low individual risk but explicit analysis would be required to confirm this conclusion~~.
of early fatality -

If this is a conclusion that can't be substantiated (I can't find any other words about this later in this document) move it to the conclusions section - If there is no real technical evaluation leave it out altogether. In any event a statement like this does not belong in the scope section -

Low power and shutdown accidents are potentially significant because the plant configuration is altered; the containment may be open and the reactor safety systems may be realigned. However, offsetting mitigating attributes include a potentially much smaller decay heat level and low pressure which allows for easier cooling of the reactor fuel. In this area, SOARCA has focused on the accidents which historically have received the most attention, the accidents initiated at full power. Also~~, one of the objectives was to provide an updated quantification~~ of ~~risk from past studies such as the 1982 Siting study and that study similarly was confined to full power reactor events.~~ If they did it wrong in the past and we are trying to update that study, we need to do it wrong now.

especially if it was an example of non-conservatism

Extreme seismic events which involve failure of the containment and lead to core damage have been excluded. We conclude that substantially more research is needed before it is feasible to undertake a realistic, best estimate analysis of such rare events. The NRC has developed plans to conduct this seismic PRA research.

Spent fuel pool accidents also contribute to overall risk associated with nuclear reactors since significant quantities of spent fuel are stored on site in such pools. Past NRC studies including the most recent publicly available study, NUREG-1738, (February 2001), "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants" would suggest that risk from the most severe spent fuel pool accidents is low yet the consequences could be serious due to the release of a large inventory of cesium and other radioisotopes. Since that time the NRC has undertaken substantial analytical and empirical research to improve both the modeling of spent fuel pool accidents as well as research to identify significant improvements to spent fuel pool safety as part of NRC's security related research following the terrorist attacks of 9/11/2001. Based on the results of this research, the NRC concludes that the estimate of spent fuel pool risk

real risk is not changed by improvements in our models - only our estimates are changed which was assessed very conservatively in past studies such as NUREG-1738, is now much lower – due to both the new physical safety improvements required by the NRC and the improved modeling capability. Therefore, the NRC when developing the SOARCA project elected to exclude spent fuel pool accidents from its scope. That does not negate the benefits of applying more detailed best estimate methods to spent fuel pool accidents at some point in the future.

that

The NRC did not include security events as part of SOARCA in order to preclude providing any specific information which may materially assist in the planning or carrying out of a terrorist attack on a nuclear power plant. However, we have stated that it was the security related studies conducted after 9/11/2001 ~~which~~ led us to conclude that previous risk studies were unnecessarily conservative in certain areas and results and that plant improvements plus improved modeling would confirm that radionuclide releases and early fatalities were substantially smaller than suggested by earlier studies.

1.5 SOARCA and Full Scope Level 3 PRA

In the selection of important sequences the SOARCA project would have ideally included those sequences found to be important to risk as demonstrated by a full-scope level 3 ~~Probabilistic Risk Assessment (~~PRA~~)~~. In practice, that was not feasible since there were no current full scope level 3 PRAs generally available, considering both internal and external events, to draw upon. However, the preponderance of level 1 PRA information, combined with our insights on severe accident behavior, is available on dominant core damage sequences, especially internal event sequences. This information, combined with our understanding of containment loadings and failure mechanisms together with radionuclide release, transport and deposition, a~~llow us to~~ utilize core damage frequency (CDF) as a surrogate criterion for risk. Thus, for SOARCA we elected to analyze sequences with a CDF greater than 10^{-6} per reactor-year. In addition, we included sequences 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. Such sequences would be associated with events involving containment bypass or leading to an early failure of the containment. By the adoption of these criteria, we are reasonably assured that the more probable and important core melt sequences will be captured.

suggests that we should

1.6 Mitigated and Unmitigated Cases

An important objective of the SOARCA project was to assess the impact of severe accident mitigative features and reactor operator actions to mitigating the accident. This was done by evaluating in detail the operator actions and equipment which may be available (including 10 CFR 50.54(hh) equipment) to mitigate the specific accident sequences to determine if time was available to take corrective action and whether the equipment itself would be available given the sequence. These mitigative measures analyses were qualitative, sequence-specific systems and operational analyses based on licensee identified mitigative measures from Emergency Operation Procedures (EOPs), Severe Accident Mitigation Guidelines (SAMGs), and other severe accident guidelines that are applicable to, and determined to be available during a scenario whose availability, capability and timing was utilized as an input into the MELCOR analyses. 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 and the likelihood of success or failure is unknown. However, the NRC has issued 10 CFR 50.54(hh) requiring plant licensees to possess the equipment, develop the strategies and have trained personnel to implement these mitigative measures. For sequences in which it was determined that mitigative measures would be taken, detailed accident progression analyses were conducted to assess the efficacy of those measures, given proper implementation. For such sequences, accident progression and offsite consequence analyses were also performed assuming the mitigative measures were not taken, in order to demonstrate the relative importance/significance of those measures.

For those scenarios within the scope of SOARCA, applicable mitigative measures that are potentially available (not eliminated by initial conditions) were identified. The systems and operations analyses were based on the initial conditions and anticipated subsequent failures to:

- verify the availability of the primary system,
- determine the availability of support systems and equipment
- determine time estimates for implementation

Based on these scenario specifications, MELCOR was used to determine the effectiveness of those mitigative measures that are expected to be available at a given time.

1.7 Key Assumptions

In the development of the accident and consequence analysis for the SOARCA project, the concepts, applications, and parameters are identified in detail in the applicable report sections. Assumptions are identified throughout the report in the appropriate sections that address the analysis. Some of the overarching assumptions used in the SOARCA project are identified below.

1.7.1 Accident Analysis

The progression of events in a severe accident has uncertainties in phenomenological responses, equipment performance, and operator actions. An independent expert panel was assembled to review the proposed base case approach as well as identify other areas of MELCOR modeling to identify areas that would benefit from improvement. The discussion of the best estimate modeling practices for MELCOR is summarized in NUREG/CR-7008 [61].

1.7.2 Consequence Analysis

Values used in the documentation provided for the sites were used whenever possible. This includes evacuation time estimates, the population surrounding the plant and other parameters as appropriate.

different font?

- Emergency response parameters were included in the modeling of each scenario for each plant.
- Cohorts (groups of population that have similar response) representing a non-evacuating population, school children, special needs population, a shadow evacuation, and the

general public in the EPZ were assigned parameters appropriate for the response attributes of the cohort.

- Ingestion dose via the food pathway was bypassed reflecting a modeling assumption that food supplies within the U.S. are sufficient so that eating contaminated food following an accident ~~is~~ unnecessary. or drinking contaminated water
- Other, non-site specific parameters were obtained from a series of studies conducted by the NRC and the Commission of European Communities to develop credible and traceable uncertainty distributions for important input parameters. The specific values used in the best estimate case were the medians of the distributions that were developed by the NRC from the results of the studies [58][59]. are

1.8 Uncertainty Analysis

As part of the SOARCA project, a number of sensitivity studies have been performed to examine issues associated with accident progression, mitigation and offsite consequences for the accident scenarios of interest. These sensitivity studies were performed to examine specific issues and to ensure the robustness of the conclusions documented in this report. Single sensitivity studies, however, do not form a complete picture of the uncertainty associated with the accident progression and offsite consequence modeling. Such a picture requires a more comprehensive evaluation of both epistemic (state of knowledge) and aleatory (random) ~~modeling~~ uncertainties.

In general terms, the "best estimate" off-site consequence results presented in this documentation reflect the aleatory uncertainty associated with weather conditions at the time of the accident scenario ~~considered~~. These best estimate off-site consequence values represent the expected (mean) value of the probability distribution obtained from a large number of weather "trials". The impact of epistemic model parameter uncertainty will be evaluated by randomly sampling distributions for model parameters that were considered to have the greatest potential impact on the off-site consequences.

Ideally, the uncertainty study described here would be conducted for each accident sequence considered by the SOARCA project. Practical considerations, including the computational expense involved, require that a more limited study be conducted. As a result, a detailed uncertainty study will be performed for a single accident sequence ~~of historical interest~~ rather than all of SOARCA sequences. In addition, to make the uncertainty study more tractable, only a subset of model parameters will be considered. The model parameters included in the study, as well as the distributions used to characterize the uncertainty in the accident progression parameters, will be determined based on the collective judgment of the SOARCA analysis team. Distributions for ~~offsite~~ consequence modeling parameters, however, will be obtained from previous work [48]. non-site specific

really? Is "historical interest" the basis we are going to use to choose?

1.9 Structure of NUREG-1935 and Supporting Documents

and a summary of the results

The structure of the NUREG is in multiple volumes. This volume is the introduction to the SOARCA project, ~~and~~ the methods and approaches used in the study. Appendix A and B contain the plant specific SOARCA results for the Peach Bottom and Surry plants, respectively. Additional plant-specific appendices will be added as volunteer-plants are identified and assessed.

2.0 ACCIDENT SCENARIO SELECTION

An accident sequence begins with the occurrence of an initiating event (for example, a loss of offsite power, a loss of coolant accident, or an earthquake) that perturbs the steady state operation of the nuclear power plant. The initiating event challenges the plant's control and safety systems, whose failure could potentially cause damage to the reactor fuel and result in the release of radioactive fission products. Because a nuclear power plant has numerous diverse and redundant safety systems, many different accident sequences are possible, depending on the type of initiating event that occurs, the amount and kind of equipment that fails, and the nature of the operator actions involved.

One way to systematically identify a complete set of possible accident sequences is to develop accident sequence logic models using event tree analysis, as is done in ~~probabilistic risk assessments (~~PRAs~~)~~. Pathways through an event tree represent accident sequences. Typically, the analysis is divided into two parts: (1) a Level 1 PRA, which represents the plant's behavior from the occurrence of an initiating event until core damage occurs, and (2) a Level 2 PRA, which represents the plant's behavior from the onset of core damage until radiological release occurs. The development of accident sequence logic models requires detailed information about the plant and the expertise of engineers and scientists from a wide variety of technical disciplines. As a result, the construction of accident sequence logic models is a complex and time consuming activity.

or is it usually until adequate core cooling can no longer be assured

Many PRAs have already been completed by the NRC and nuclear power plant licensees. However, due to the dynamic improvements in PRA technology and plant capabilities and performance, the more current PRA information available was given more importance. The following sections describe the overall approach used to select accident scenarios for further analysis, and the results of the accident scenario selection process.

2.1 Approach

Figure 1 only uses the word sequence — Either change the figure or explain sequence scenario differences-

Figure 1 illustrates the overall process used to identify and characterize accident scenarios for the SOARCA project. The SOARCA scenarios were selected from the results of existing PRAs. Some of these existing PRAs model accident sequences out to the point of radiological release (i.e., they are Level 2 PRAs); however, the majority of existing PRAs are limited to the onset of core damage (i.e., Level 1 PRAs). Therefore, the SOARCA scenario selection process was developed with an eye towards the type and limitations of the information contained in existing PRAs. Core-damage sequences from previous staff and licensee PRAs were identified and binned into core-damage groups. A core-damage group consists of core-damage sequences that have similar characteristics with respect to severe accident progression (timing of important phenomena) and containment or engineered safety feature operability. The groups were screened according to their approximate core-damage frequencies to identify the most significant groups. Finally, the accident scenario descriptions were augmented by assessing the status of containment systems (which are not typically modeled in Level 1 PRAs).

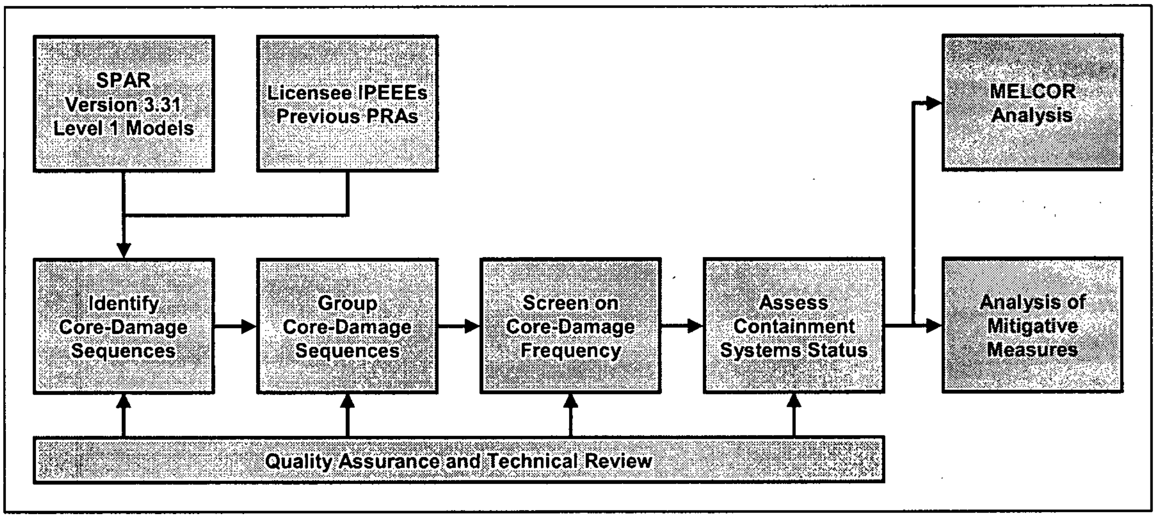


Figure 1 SOARCA Accident Scenario Selection Process

The scope of the analyses with MELCOR and MACCS2 were limited to sequence groups with core damage frequencies (CDFs) greater than 1×10^{-6}/year. For sequence groups involving containment bypass, sequence groups selected for analysis were those with CDFs greater than 1×10^{-7}/year because of the potential for bypass sequence groups to have higher consequences and higher risk. To accomplish this, the release characteristics were grouped so that they are representative of scenarios binned into those groups and the groups are sufficiently broad to include the potentially risk-significant but lower-frequency scenarios. As a result of limitations in available Level 2 analyses and models, the scenario selection and screening was performed using a core-damage frequency (CDF) of 10^{-6} per reactor-year screening value rather than the radionuclide release frequency.

and for

In order to account for uncertainties, known limitations in the plant PRA models, and to partially limit the potential for screening out risk-significant (higher consequence) scenarios due to screening at the core-damage level, the following screening guidelines were established:

Just said this in the preceding paragraph!

- 10^{-6} per reactor-year for most scenarios, and
- 10^{-7} per reactor year for scenarios that are known to have the potential for higher consequences (e.g., containment bypass scenarios such as steam generator tube rupture and interfacing system loss-of-coolant accident (ISLOCA) initiators).

Certain initiating events have the potential to affect multiple units at a given site or multiple sites. For example, on August 14, 2003, a widespread loss of the Nation's electrical power grid (blackout) resulted in loss of offsite power events at nine U.S. commercial nuclear power plants located at six different sites. An earthquake could similarly affect multiple units. Most PRAs developed to date do not explicitly consider multi-unit accidents because current NRC policy is to apply the Commission's Safety Goals (51 FR 28044) [6] and subsidiary risk acceptance guidelines (see Regulatory Guide 1.174 [7]) on a "per reactor" basis. Therefore, no multi-unit accident scenarios were selected for the SOARCA project. However, the possible need to assess multi-unit risks has been referred to the NRC's Generic Issues Program for resolution.

don't need review the same page

~~Containment systems availabilities were assessed using system dependency tables which delineate the support systems required for performance of the target front line systems and from a review of existing system fault trees.~~ In addition, for external event initiated scenarios judgments were made on system availability based on the type and severity of the external event initiating event (e.g., large seismic event).

2.2 Scenarios Initiated by Internal Events

The scenarios generated by internal events and the availability of containment systems for these scenarios were identified using the NRC's plant-specific standardized plant analysis risk (SPAR) models, licensee PRAs, and other risk information sources. The SPAR models are a set of 76 linked fault tree/event tree models used by the NRC to evaluate the risk of operations at all 104 U.S. nuclear power plants. The detail of a SPAR model approaches that of a typical utility PRA with approximately 1200 basic events, 150 fault trees, and 15 event trees. The core-damage frequencies calculated from the current SPAR models are similar to that calculated in utility PRAs. The SPAR model accident sequence results (minimal cutsets) were also compared to the results from utility PRAs. The SPAR models are developed and maintained under a formal quality assurance program by the NRC but are not publically available.

The following process was used to determine the scenarios for further analyses in the SOARCA project:

1. Candidate accident scenarios were identified in analyses using plant-specific, SPAR models (Version 3.31).

    a. Initial Screening – Screened out initiating events with low CDFs ($<10^{-7}$) and sequences with a CDF $<10^{-8}$. This step eliminated 4% of the overall CDF for Peach Bottom and 7% of the overall CDF for Surry.

    b. Sequence Evaluation – Identified and evaluated the dominant cutsets for the remaining sequences. Determined system and equipment availabilities and accident sequence timing.

    c. Scenario Grouping – Grouped sequences with similar times to core damage and equipment availabilities into scenarios.

2. Containment systems availabilities for each scenario were assessed using system dependency tables which delineate the support systems required for performance of the target front-line systems and from a review of existing SPAR model system fault trees.

3. Core-damage sequences from the licensee PRA model were reviewed and compared with the scenarios determined by using the SPAR models. Differences were resolved during meetings with licensee staff.

4. The screening criteria (CDF $< 10^{-6}$ for most scenarios, and $< 10^{-7}$ for containment bypass sequences) were applied to eliminate scenarios from further analyses.

This process provides the basic characteristics of each scenario. However, it is necessary to have more detailed information about scenario than is contained in a PRA model. To capture the additional scenario details, further analysis of system descriptions and review of the normal and emergency operating procedures (EOPs) is required. This review includes the analysis of mitigation measures beyond those treated in current PRA models. Mitigation measures treated in SOARCA include the licensee's EOPs, severe accident management guidelines (SAMGs), and 10 CFR 50.54(hh) mitigation measures. Section 3.0 describes the mitigation measures assessment process used to determine what mitigation measures would be available and the associated timing to implement.

2.2.1 Scenarios Initiated by External Events

Detailed sequence characteristics are more difficult to specify for scenarios initiated by external events (e.g., fire, seismic, flooding) due to the lack of external event PRA models industry-wide. The external event scenarios selected for analysis in the SOARCA project are representative of those that might arise due to seismic, fire or internal flooding initiators. Although they were derived from a review of past studies such as ~~the~~ NUREG-1150, individual plant examination for external event (IPEEE) submittals, and other relevant generic information, they do not represent specific accident sequences from any of these prior studies.[1]

In order to specify the scenarios for further analysis and the assessment of mitigative measures, the selected scenarios were assumed to be seismically initiated since, in general, seismic-initiated scenarios are the most restrictive in terms of the ability to successfully implement onsite mitigative measures and offsite protective actions. In addition, the seismic-initiated scenarios were judged to be important contributors to the external event core damage and release frequencies.

reference?

Various data sources and assessments readily available are examined to identify dominant external event sequences that may be potentially applicable to a specific NPP. Dominant sequence information from these sources is captured in a report. These sequences are examined and compared to select a set of representative dominant external event sequences that are deemed to be applicable to the plant. Representative dominant external event sequence descriptions, containment safeguards status, and a sequence frequency estimate are provided whenever possible.

past, rather than present

Did we do this?

No attempt is made to match the selected representative dominant external event sequences to actual sequence frequencies from one source; nor is any criterion used to capture a certain percentage of total external events CDF. General insights from the literature are also used to select representative sequences. Care is taken that this selection maintains the (perceived) relative importance of external events CDF versus internal events CDF. The fact that some of

[1] "External events" in this document refer to all other events at-power than those modeled routinely as internal events in a SPAR model. External events include internal flooding and fire, seismic events, extreme wind, tornado and hurricane related events, and other events that may be applicable to a specific site. The assessment is based on readily available information to NRC/RES analysts at the time of preparation of this report (such as NUREGs, SPAR-EE models, IPE and IPEEE submittals). The nature, vintage, and variety of the information ~~may require~~ required a quantitative evaluation, supplemented with a suggested CDF for a representative dominant sequence.

? Past instead of future!

the information available is dated and is already superseded (such as seismic hazard curves, internal fire frequencies, and methodology, internal flooding analyses) is taken into account to avoid undue conservatisms.

2.3 Accident Scenarios Selected for Surry

Four accident sequences were selected for the Surry plant (two initiated by internal events, and two initiated by external events). The following sections identify each selected accident scenario, provide its representative core-damage frequency, and summarize the accident scenario in terms of its initiating event, equipment failures and operator errors.

2.3.1 Surry Internal Event Scenarios

Two internal event scenarios for Surry were determined to meet the criteria for further analysis.

always put both SPAR and Licensee values, especially when we choose one over the other — Indicate w/ ?, say, * which one so ARCA is based on —

1. *Initiating Event:* Spontaneous Steam Generator Tube Rupture

   Representative CDF: 5×10^{-7} per reactor-year (SPAR)

fail to

<u>Scenario Summary:</u> This scenario is initiated by a spontaneous rupture in one steam generator tube. The operators fail to isolate the faulted steam generator, ~~to~~ cooldown and depressurize the reactor, and fail to initiate long term heat removal. Core damage occurs due to refueling water storage tank (RWST) depletion and the operator failure to refill the RWST or cross-connect to another water source. High pressure injection (HPI), low pressure injection (LPI) and containment spray are available, if needed. However, high pressure recirculation, low pressure recirculation and the recirculation sprays will be unavailable as a result of lack of water in the containment sump. Auxiliary feedwater (AFW) is available; however the timing, flow rates and into which steam generators AFW injection occurs cannot be determined from the systems analysis.

<u>Comparison with Licensee PRA:</u> The licensee PRA calculates a CDF 1×10^{-6} per reactor-year for this scenario. The conditional core-damage probabilities are virtually identical for the SPAR analysis (1.4×10^{-4}) and for the licensee PRA (1.5×10^{-4}). The difference in the calculated CDFs is mainly attributable to the difference in initiating event frequency. Since both the SPAR model and licensee calculated CDFs for this scenario are above the 1×10^{-7} per reactor-year threshold for containment bypass scenarios; this scenario was retained for further analysis.

2. *Initiating Event:* Interfacing Systems LOCA in the LPI System

   Representative Frequency: 7×10^{-7} per reactor-year (licensee PRA)

<u>Scenario Summary:</u> This scenario is initiated by failure of two check valves in series in the discharge path of the LPI system. The flow of reactor coolant system (RCS) fluid passing through the failed check valves results in pressurization of the low-pressure LPI piping in the Safeguards Building and its subsequent pipe rupture. The rupture location cannot be isolated. The ability to inject via the LPI is failed by the rupture. The HPI system remains available

because the pumps are located in a separate location. Core damage occurs due to RWST depletion and operator failure to refill the RWST or cross-connect to another water source.

Comparison with Licensee PRA: The SPAR model for Surry calculates a CDF 3×10^{-8} per reactor-year for this scenario. The largest contributor to the difference in scenario CDFs between the licensee PRA and SPAR model is the conditional probability of the low-pressure portion of the LPI piping system rupturing given that the two isolation check valves have failed open. The SPAR model assigns a value of 0.1 for this event while the Surry PRA assigns a probability of failure of 1.0. This sequence group was retained for further analysis because the licensee PRA frequency exceeds the SOARCA screening criteria and it has historically been important to PWR risk.

2.3.2 Surry External Event Scenarios

Two external event scenarios for Surry were determined to meet the criteria for further analysis.

1. *Initiating Event:* Seismic-Initiated Long-Term Station Blackout

   Representative Frequency: 1×10^{-5} to 2×10^{-5} per reactor-year

Scenario Summary: This scenario is initiated by a moderately large earthquake (0.3–0.5 peak ground acceleration - PGA). The seismic event results in loss of offsite power (LOOP) and failure of onsite emergency alternating current (AC) power resulting in a station blackout (SBO) event where neither onsite nor offsite AC power are recoverable. All systems dependent on AC power are unavailable, including the containment systems (containment spray and fan coolers). The turbine-driven auxiliary feedwater (TDAFW) system is available initially. In the long term loss of the TDAFW may occur due to battery depletion and loss of direct current (DC) power for sensing and control. Due to loss of pump seal cooling, a reactor coolant pump seal LOCA may occur.

2. *Initiating Event:* Seismic-Initiated Short-Term Station Blackout

   Representative Frequency: 1×10^{-6} to 2×10^{-6} per reactor-year

Scenario Summary: This scenario is initiated by a large earthquake (0.5–1.0 PGA). The seismic event results in a LOOP and failure of onsite emergency AC power resulting in a SBO event where neither onsite nor offsite AC power are recoverable. All systems dependent on AC power are unavailable, including the containment systems (containment spray and fan coolers). The seismic event also results in a loss of DC power resulting in the unavailability of the TDAFW system.

2.4 Accident Scenarios Selected for Peach Bottom

Two accident scenarios were selected for the Peach Bottom plant (both initiated by a seismic event). The following sections identify each selected accident scenario, provide its representative core-damage frequency, and summarize the accident scenario in terms of its initiating event, equipment failures and operator errors.

2.4.1 Peach Bottom Internal Event Scenarios

No internal event scenarios for Peach Bottom met the criteria for further evaluation.

2.4.2 Peach Bottom External Event Scenarios

1. *Initiating Event:* Seismic-Initiated Long-Term Station Blackout

   Representative Frequency: 1×10^{-6} to 5×10^{-6} per reactor-year

Scenario Summary: This scenario is initiated by a moderately large earthquake (0.3–0.5 PGA). The seismic event results in a LOOP, failure of onsite emergency AC power and failure of the Conowingo Dam power line resulting in a SBO event where neither onsite nor offsite AC power are recoverable. All systems dependent on AC power are unavailable, including the containment systems (containment spray). The turbine-driven injection systems, high pressure coolant injection (HPCI) and/or reactor core isolation cooling (RCIC), are available initially. Loss of room cooling and/or battery depletion results in eventual failure of these systems leading to core damage.

2. *Initiating Event*: Seismic-Initiated Short-Term Station Blackout

   Representative Frequency: 1×10^{-7} to 5×10^{-7} per reactor-year

Scenario Summary: This scenario is initiated by a large earthquake (0.5–1.0 PGA). The seismic event results in a LOOP, failure of onsite emergency AC power and failure of the Conowingo Dam power line resulting in a SBO event where neither onsite nor offsite AC power are recoverable. All systems dependent on AC power are unavailable, including the containment systems (containment spray). In addition, HPCI and RCIC are unavailable due to loss of DC power.

Note: The short term station blackout scenario does not meet the SOARCA screening criterion of 1×10^{-6} per reactor-year; however, the scenario was retained for analysis in order to assess the risk importance of a lower frequency, higher consequence scenario. This type of scenario has been a risk-important severe accident scenario in past PRA studies and, at a frequency of 5×10^{-7} per reactor-year; it is only a factor of two below the screening criterion.

2.5 Generic Factors

core damage frequencies for

The results of existing PRAs indicate that the likelihood of a nuclear power plant accident sequence that releases a significant amount of radioactivity is very small due to the diverse and redundant barriers and numerous safety systems in the plant; the training and skills of the reactor operators; testing and maintenance activities; and the regulatory requirements and oversight of the NRC. In addition, it is important to recognize that ~~risk estimates of~~ nuclear power plants have decreased over the years. There are several reasons for these decreases:

- Utilities have completed plant modifications intended to remedy concerns raised in earlier PRAs.

- Plants exhibit better performance, as evidenced by reductions in initiating event frequencies, improvements in equipment reliability, and higher equipment availability. Nuclear power plant equipment has become more reliable and available due to improved maintenance practices motivated by implementation of the Maintenance Rule (10 CFR 50.65) [8].

- New regulations have been created, such as the ATWS Rule (10 CFR 50.62) [9] and the Station Blackout Rule (10 CFR 50.63) [10], which directly affect the likelihood of certain types of accidents.

- PRA methodologies have improved, allowing a more realistic assessment of risk to be made. In this category, improvements in common-cause failures analysis and human reliability analysis are noteworthy. (human reliability analyses have not been completed for the 10CFR 50.54(hh) measures have not been performed)

As a result, ~~risk~~ estimates are dynamic in the sense that they reflect the impacts of constantly changing plant operational, regulatory, and PRA technology environments. Any attempt to identify significant accident sequences should be viewed as a "snapshot" of the plant at the time the analysis was completed.

core damage frequency.

3.0 METHODS USED FOR MITIGATIVE MEASURES ASSESSMENT

A fundamental objective of SOARCA is to develop state-of-the-art analyses of accident progression, radiological release, and offsite radiological health consequences for risk-important severe accident sequence groups. Included within this objective is to provide insight into the effectiveness and benefits of mitigation measures currently employed at operating reactors. Section 2.0 describes the PRA information sources including the NRC's SPAR models, licensees' PRA models, NUREG-1150 and additional expert judgment that were used to identify risk-important sequence groups leading to core damage and containment failure or bypass. This section describes the methods used to determine what mitigation measures would be available and the associated timing to implement. This includes mitigation measures beyond those treated in current PRA models. Mitigation measures treated in SOARCA include the licensee's emergency operating procedures (EOPs), severe accident management guidelines (SAMGs), and 10 CFR 50.54(hh) mitigation measures. 10 CFR 50.54(hh) mitigation measures refer to additional equipment and strategies required by the NRC following the terrorist attacks of September 11, 2001, to further improve mitigation capability. It is expected that these measures would be implemented by the licensee's emergency response organization in accordance with the approved emergency plan. Further,

all? eg EOP SAMG B.5.b or to give [illegible] B.5.b?

3.1 Site-Specific Mitigation Strategies

In preparation for the detailed, realistic modeling of accident progression and offsite consequences, the SOARCA project staff had extensive cooperation from the licensees to develop high fidelity plant systems models, define operator actions including the most recently developed mitigative actions, and develop models for simulation of site-specific and scenario-specific emergency planning. Further, in addition to input for model development, licensees provided information from their own PRA on accident scenarios. Through table-top exercises (with senior reactor operators, PRA analysts, and other licensee staff) of the selected scenarios, licensees provided input on the timing and nature of the operator actions to mitigate the selected scenarios. The licensee input for each scenario was used to develop timelines of operator actions and equipment lineup or setup times for the implementation of the available mitigation measures. This includes mitigation measures beyond those treated in current PRA models.

~~Mitigation measures treated in SOARCA include Emergency Operating Procedures (EOPs), Severe Accident Management Guidelines (SAMGs), and 10 CFR 50.54(hh) mitigation measures. 10 CFR 50.54(hh) mitigation measures refer to additional equipment and strategies required by the NRC following the terrorist attacks of September 11, 2001, to further improve severe accident mitigation capability.~~ NRC inspectors completed the verification of licensee implementation (i.e., equipment, procedures, and training) of 10 CFR 50.54(hh) mitigation measures in December, 2008.

Scenarios identified in SOARCA included both externally and internally initiated events. The externally initiated events frequently included events for which seismic, fire, and flooding initiators were grouped together. For the externally initiated events, the timeline of operator actions was developed assuming the initiator was a seismic event because the seismic initiator was judged to be the most severe initiator in terms of timing and with respect to how much equipment would be available to mitigate. Thus, there is some conservatism in attributing all of the event likelihood to a seismic initiator.

3.1.1 Sequence Groups Initiated by External Events

The PRA screening identified the following sequence groups that were initiated by external events and met the SOARCA screening criteria of $1x10^{-6}$/reactor-year for containment failure events and $1x10^{-7}$/reactor-year for containment bypass events:

- Peach Bottom long-term station blackout – $1x10^{-6}$ to $5x10^{-6}$/reactor-year
- Surry long-term station blackout – $1x10^{-5}$ to $2x10^{-5}$/reactor-year
- Surry short-term station blackout – $1x10^{-6}$ to $2x10^{-6}$/reactor-year
- Surry short-term station blackout with thermally induced steam generator tube rupture – $1x10^{-7}$ to $8x10^{-7}$/reactor year

~~These sequence groups were initiated by a seismic, fire, or flooding event. The mitigation measures assessment for each of these sequence groups was performed assuming the initiator was a seismic event, because it was judged to be limiting in terms of how much equipment would be available to mitigate. Fewer mitigation measures are expected to be available for a seismic event than for an internal fire or flooding event~~. For these sequence groups, the seismic PRAs provided information on the initial availability of installed systems. Based on the estimated level of plant damage, the availability of 10 CFR 50.54(hh) mitigation measures, their implementation time, and the timing and effectiveness of the emergency response organization support (e.g., in the Technical Support Center and Emergency Operating Facility) was evaluated.

It is important to note that, although it is not included in the above list, the seismically induced Peach Bottom short-term station blackout was also retained for analysis. With a frequency of $1x10^{-7}$ to $5x10^{-7}$/reactor year this scenario does not explicitly meet the SOARCA screening criterion. Nonetheless, it was retained in order to assess the risk importance of a lower frequency, higher consequence scenario.

ac power installed ac & dc power

Seismic events considered in SOARCA result in loss of offsite and onsite AC power, and, for the more severe seismic events, loss of DC power. Under these conditions, the turbine-driven systems RCIC and TD-AFW are important mitigation measures. BWR SAMGs include starting RCIC without ~~electricity~~ to cope with station blackout conditions. This is known as RCIC black start. 10 CFR 50.54(hh) mitigation measures have taken this a step further and also include long-term operation of RCIC without ~~electricity~~ (RCIC black run), using a portable generator to supply indications such as reactor pressure vessel level indication to allow the operator to manually adjust RCIC flow to prevent RPV overfill and flooding of the RCIC turbine. Similar procedures have been developed for PWRs for TD-AFW. For the Peach Bottom and Surry long-term station blackout sequence groups, RCIC and TD-AFW can be used to cool the core until battery exhaustion. After battery exhaustion, black run of RCIC and TD-AFW can be used to continue to cool the core. MELCOR calculations are used to demonstrate core cooling under these conditions.

Seismic PRAs for Peach Bottom and Surry do not describe general plant damage and accessibility. The damage was assumed to be widespread and accessibility to be difficult, consistent with the unavailability of many plant systems. The seismic event was assumed to fail the Condensate Storage Tank in the Peach Bottom long-term station blackout, which is the primary water reservoir for RCIC. Consequently, RCIC must be initially supplied from the torus. MELCOR calculations showed that several hours would be available before torus temperature and pressure conditions precluded this. However, this would provide sufficient time to identify or arrange for another water reservoir for RCIC, such as the Peach Bottom cooling tower basin (a large low lying reinforced concrete structure). For the Surry long-term station blackout, the TD-AFW system and the Emergency Condensate Storage Tank (ECST) were not expected to fail. Consequently, the cooling water was supplied to the steam generators for RCS heat removal. It was assumed that eventually operators would provide make-up to the ECST. For the Surry short-term station blackout, the ECST was assumed to fail and an alternative reservoir was assumed to be available by 8 hours; this could be achieved by using a fire truck or portable pump to draw from the river.

Also, for the Surry short-term station blackout, the low pressure injection and containment spray safety-related piping were judged not likely to fail. This judgment was primarily based on NUREG/CR-4334, "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants" [11], to help extrapolate the potential viability of safety-related piping after a 1.0 g event. This conclusion also considered other related studies including a German study that physically simulated ground motion equal to 1 g on an existing plant. The integrity of this piping provided a connection point for a portable, diesel-driven pump to inject into the RCS or into the containment spray systems. Licensee staff estimated that transporting the pump and connecting it to plant piping takes about two hours. However, for the short-term station blackout this mitigation measure was estimated to take 8 hours due to the higher level of damage. Since the installation time was beyond the estimated time to fuel damage and vessel failure (3 hours to core damage, 7 hours to lower head failure), the containment spray system was the preferred mitigation measure. A better understanding of the effect of large seismic events on general plant conditions would be helpful in reducing uncertainty in availability and accessibility for mitigation measures.

10 CFR 50.54(hh) mitigation measures include portable equipment such as portable power supplies to supply indication, portable diesel-driven pumps, and portable air bottles to open air-operated valves, together with procedures to implement these measures under severe accident conditions. At the time of the Surry site visit, the licensee had their portable equipment and the site fire truck onsite in a structure away from the containment. Hence, it was believed that portable equipment could be accessed and deployed for the seismic conditions evaluated in SOARCA. At the time of the Peach Bottom site visit, the licensee had not procured the required portable equipment.

Time estimates to implement individual mitigation measures were provided by licensee staff for each sequence group based on scenario descriptions provided by the NRC. The time estimates take into account the plant conditions following the seismic event. Also, for portable equipment at Surry, the time estimates reflect exercises run by licensee staff that provided actual times to move the equipment into place. The time estimates for staffing the Technical Support Centers and the Emergency Operating Facilities also were provided by licensee staff and reflect the possible effect of the seismic event on roads and bridges.

The mitigation measures assessment noted the possibility of bringing in equipment from offsite (e.g., fire trucks, pumps and power supplies from sister plants or from contractors), but it did not quantify the types, amounts, and timing of this equipment arriving and being implemented. Additional information on equipment available offsite and time estimates for transporting this equipment is available in Section 3.2.

Since multi-unit accident sequences were NOT ? selected for the SOARCA project, the mitigation measures assessment for external events was performed assuming that the operators only had to mitigate an accident at one reactor, even though Peach Bottom and Surry are two-unit sites.

3.1.2 Sequence Groups Initiated by Internal Events

The PRA screening identified the following sequence groups that were initiated by internal events and met the SOARCA screening criteria of 1×10^{-6}/reactor-year for containment failure events and 1×10^{-7}/reactor-year for containment bypass events:

- Surry interfacing systems LOCA – 7×10^{-7}/reactor-year (licensee PRA), 3×10^{-8}/reactor-year (SPAR)
- Surry spontaneous steam generator tube rupture – 5×10^{-7}/reactor-year

These sequence groups result in core damage as a result of assumed operator errors. For the interfacing systems LOCA, the operators fail to refill the RWST or cross-connect to the unaffected unit's RWST. For the spontaneous SGTR, the operators fail to 1) isolate the faulted SG, 2) depressurize and cooldown the RCS, and 3) refill the RWST or cross-connect to the unaffected unit's RWST.

The SPAR model and the licensee's PRA concluded that these two events proceed to core damage as a result of the above postulated operator errors. However, these PRA models do not appear to have credited the significant time available for the operators to correctly respond to events. They also do not appear to credit technical assistance from the TSC and the EOF. For the ISLOCA, the realistic analysis of thermal hydraulics presented in Appendix B subsequently estimated 3 hours until the RWST is empty and 10 hours until fission product release begins, providing time for the operators to correctly respond. The ISLOCA time estimates are based on a double ended pipe rupture. These estimates could be longer for smaller break sizes. For the SGTR, the realistic analysis of thermal hydraulics showed from 24 to 48 hours until core damage begins. Therefore, based on realistic time estimates by which the technical assistance is received from the TSC and the EOF, it was highly likely the operators would correctly respond to the events. These time estimates included consideration of indications that the operators would have of the bypass accident, operator training on plant procedures for dealing with bypass accidents and related drills, and assistance from the TSC and EOF, which were estimated to be staffed and operational by 1 to 1.5 hours into the event.

The mitigation measures assessment for internal events also included 10 CFR 50.54(hh) mitigation measures, but these measures were subsequently shown to be redundant to the wide variety of equipment and indications available for mitigating the ISLOCA and SGTR. ISLOCA and SGTR are internal events that involve few equipment failures and are controlled by operator errors.

The PRA screening for Peach Bottom initially identified the Loss of Vital AC Bus E12 sequence group as exceeding the SOARCA screening criterion of 1×10^{-6}/reactor-year. However, an inappropriate modeling assumption was subsequently found in the SPAR model, and the sequence group frequency was determined to be below the SOARCA screening criterion. However, by the time the issue was discovered, the mitigation measures assessment and the MELCOR analysis were complete. The MELCOR analysis described in Appendix A demonstrated that this sequence group did not result in core damage, even without crediting 10CFR50.54(hh) mitigation measures, contrary to the more conservative treatment in SPAR. Nevertheless, the mitigation measures assessment and the MELCOR analysis for this sequence are described in this report to demonstrate the benefit of a detailed review of success criteria using integrated thermal-hydraulic analysis.

3.2 Unmitigated Scenarios - Truncation of Releases

The assessment of mitigation measures, including emergency operating procedures, severe accident management guidelines, and security-related mitigation measures, led the project staff to conclude that all of the identified severe accident scenarios could be suppressed or mitigated if implementation of these procedures were successful. To quantify the benefits of the mitigation measures ~~and to provide a basis for comparison to past analyses of unmitigated severe accident scenarios,~~ the project staff also analyzed the scenarios ~~conservatively~~ assuming the events proceed as unmitigated by available onsite mitigation measures and lead ultimately to an offsite release. This NUREG refers to these as unmitigated scenarios, because they are not effectively mitigated in the short-term by onsite resources. For these unmitigated scenarios, the project staff performed analysis to estimate the time by which offsite resources would be brought onsite to control or stop the long-term revaporization release of fission products from the containment and other plant buildings.

There are a multitude of resources available at the state, regional and national level that would be available to mitigate a NPP accident. The staff reviewed available resources and emergency plans and determined that adequate mitigation measures could be obtained within 24 hours and fully implemented within 48 hours.

The National Response Framework (NRF) would be implemented in response to a severe nuclear power plant accident to coordinate the national level response. Under the NRF, Department of Homeland Security (DHS) would be the coordinating agency and NRC would be a cooperating agency. The NRF is exercised periodically and provides access to the full resources of the Federal government. The NRC has an extensive, well-trained and exercised emergency response capability and has onsite Resident Inspectors. These onsite inspectors are equipped and available to provide first-hand knowledge of accident conditions. The NRC would activate the incident response team at the NRC regional office and headquarters. The focus of the NRC response is to ensure that public health and safety is protected and to assist the licensee with the response by working with DHS to coordinate the national response. Concurrently, the NRC regional office would send a site team to staff positions in the reactor control room, Technical Support Center and Emergency Operations Facility to support the response. The NRC performs an independent assessment of the actions taken or proposed by the licensee to confirm such actions will ~~stop core damage~~ lead eventually to improved plant conditions.

Both Surry and Peach Bottom are supported by a remote EOF. The emergency response organization at the EOF has access to fleet wide emergency response personnel and equipment including the 10 CFR 50.54(hh) mitigation measures and equipment from sister plants. These assets as well as those from neighboring utilities and state preparedness programs could be brought to bear on the accident if needed. Every licensee participates in ~~a~~ full onsite and offsite exercises biannually where response to severe accidents and coordination with offsite response organizations is demonstrated and inspected by NRC and the Federal Emergency Management Agency (FEMA). In addition the Institute for Nuclear Power Operations and the Nuclear Energy Institute would activate their emergency response centers to assist the site as needed.

All of the described resources would be available to the site to mitigate the accident. While some of these efforts would be ad hoc, knowledgeable personnel and an extensive array of equipment would be available and were considered in the conclusion that radiological releases would be truncated within 48 hours.

4.0 SOURCE TERM ANALYSIS

The source term is defined [for purposes of this document] as the quantity, timing, and characteristics of the release of radioactive material to the environment following a postulated severe accident. The NRC has defined, calculated, and used source terms for a variety of research and regulatory activities. Two uses include (a) siting and regulatory applications and (b) probabilistic risk or consequence assessments. Two source terms used for regulatory applications include TID-14844 [12] and the alternate source term [13]. In contrast to the definition above, the regulatory source terms are releases to the containment, which are available for release to environment. The second use of the source term is an assessment of health consequence risks from severe accidents. There are many significant examples of the latter application including the Sandia Siting study [4], NUREG-1150 [5], and SOARCA. In the Sandia Siting Study, ~~the postulated frequency distribution of~~ five prescribed source terms of increasing severity were defined and used to calculate ~~the~~ health consequences ~~and risk~~. In NUREG-1150, a comprehensive, plant-specific evaluation of all significant source terms ~~were~~ [was] calculated using event tree models ~~and~~ [which were] then grouped into a much larger but manageable number of characteristic source terms to calculate ~~the~~ health consequences and risk. In the present SOARCA study, individual scenario source terms are evaluated using MELCOR code calculations and then evaluated for health consequences.

Some background in key studies for regulatory and probabilistic applications is described in Section 4.1 below. Figure 2 shows a timeline of key events and NRC studies in the evolution of nuclear safety technology. The key source terms studies cited in the timeline that preceded the SOARCA program are shown in the figure and discussed in Section 4.1 below. Next, a history of the severe accident source term codes developed by the NRC is described in Section 4.2. The MELCOR code is the culmination of the NRC research and code development ~~of~~ severe accident phenomena for source term evaluations. The scope of the MELCOR code and ~~the relevant experimental programs supporting~~ its advanced modeling capabilities are summarized in Section 4.3. The MELCOR modeling approach used in [into] the SOARCA analyses is [for] presented in Section 4.4. The MELCOR modeling approach includes the development of the plant models, the best practices approaches to important but uncertain phenomena and equipment performance, recent advances in source term models, and the methods used to calculate the radionuclide inventories.

we moved this info the to Melcor NUREG/CR

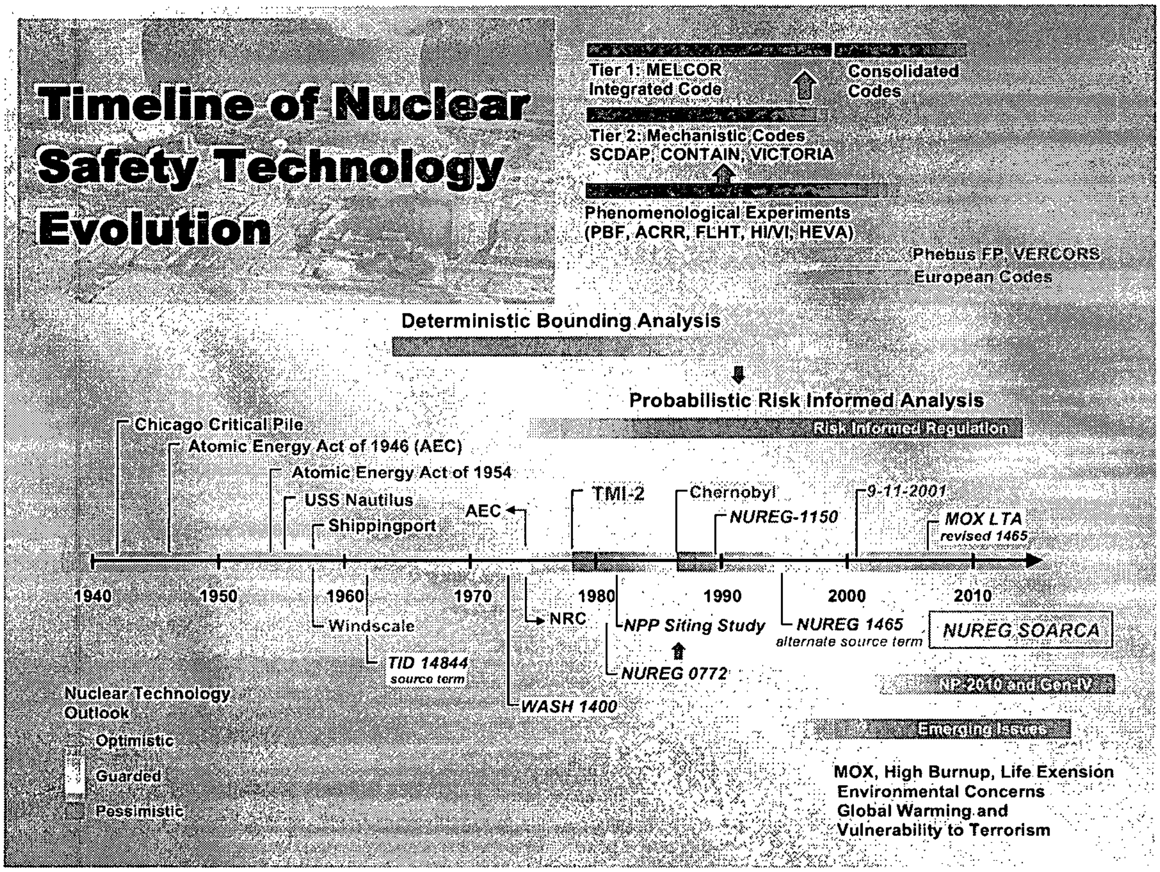


Figure 2 Timeline of Key Nuclear Power Events and Safety Studies.

4.1 Source Term Study Background

One of the earliest estimates of the source term came from the WASH-740 study in 1957 [1]. Three core damage cases were defined with increasing levels of severity. The first case was defined as a situation in which there was a major damage to the core resulting in failure of the vessel. However, the containment remained intact, thus preventing a major release of radioactivity to the environment. This case was subsequently used to define the characteristics of the source term for reactor siting, i.e., TID-14844 [12]. In the other two cases, there were releases offsite.

The TID-14844 source term postulated the release of all the noble gases, 50% of the iodine, and 1% of the radioactive solids to the containment. In addition, TID-14844 provided assumptions for containment leakage and for atmospheric transport. However, it was recognized that the procedures and results specified in TID-14844 were approximations, sometimes relatively poor ones, to the results which would be obtained if the effects of the all influencing variables could be recognized and associated with fixed levels of uncertainty – an impossibility in the state-of-the-art at the time [14]. Nevertheless, TID-14844 was codified as "the maximum credible accident" in the siting regulations of 10 CFR Part 100, "Reactor Site Criteria" [15].

The next most significant source term study, the Reactor Safety Study (WASH-1400) [3], was the first systematic attempt to provide realistic estimates of public risk from potential accidents in commercial nuclear power plants. The 1975 study included analytical methods for determining both the probabilities and consequences of various accident scenarios. Event trees and fault trees were used to define important accident sequences and to quantify the reliability of engineered safety systems. A ~~more~~ comprehensive list of nine PWR and five BWR source terms was developed. All the accidents that were believed to contribute significantly to the overall core melt frequency were grouped, or "binned," into the source term categories. The WASH-1400 source terms included characterizations of accident timing, the release duration (e.g., puff or sustained release), and the energy of the release for plume loft considerations. As an improvement over TID-14844, the radioactivity was described using eight chemical categories. The 54 ~~most health-significant~~ isotopes were used in health consequence calculations. those

The WASH-1400 methodology ~~used to predict~~ the health effects from the source term was based on the newly developed Calculation of Reactor ~~Accident Consequences~~ (CRAC) code ~~[16]~~, which calculated the atmospheric dispersion and health consequences. However, an integrated tool for the ~~calculation of the source term~~ did not exist. The estimation of the source term used the best analytic procedures available at the time. When ample data was available, a model for the phenomenon was included as realistically as possible, but when data were lacking, consideration of the phenomenon was omitted. The resultant source terms reflected uncertainties and poor understanding of applicable phenomena. Uncertainties in accident frequencies were accounted for by adding 10% of the likelihood of each release category into the next larger and the next smaller category.

Subsequently, the NRC documented the technical basis for source terms in NUREG-0772 [17]. NUREG-0772 assessed the advancements assumptions, procedures, and available data for predicting fission product behavior. Four ~~conclusions~~ of the NUREG-0772 study were (1) a new definition of the chemical form of iodine (i.e., CsI was the dominant form), (2) the potential retention of CsI within the vessel or containment versus elemental iodine, (3) the inclusion of in-vessel retention, and (4) the role of containment engineering safety features (e.g., sprays, suppression pools, and ice condensers). However, much of the quantitative assessment in NUREG-0772 was based on scoping calculations that were only applicable to specific conditions. In particular, the examination of fission product behavior in different regions of the plant with different accidents was conducted in parallel with limited consideration of integral effects. The potential impact of the NUREG-0772 findings on reactor regulation was also examined and the results were documented in NUREG-0771 [18].

NUREG-0771 and NUREG-0772 ~~studies~~ formed the basis for the designation of five accident groups as being representative of the spectrum of potential accident conditions, which were documented in NUREG-0773 [19]. In 1982, the Sandia Siting Study [4] was performed using the NUREG-0773 source terms. The five source terms were assessed to adequately span the range of possible source terms. The source terms were developed from separate effects computer code analyses that were performed in 1978. The source terms were used to calculate accident consequences at 91 United States reactor sites using site-specific population data and a

Page 5 says 92 check for all

mixture of site-specific and regionally–specific meteorological data. An objective of the SOARCA study is to update this study.

In response to emerging severe accident research technology and computing power, a study was performed at Battelle Columbus Laboratories that involved the development and modification of a number of separate effects severe accident computer codes based on emerging severe accident research. The codes were coupled together to form a code suite that could calculate a complete accident sequence. The source terms for about 25 specific sequences were calculated for five operating plants using the new Source Term Code Package (STCP) code [20]. While the STCP was a significant step forward in deterministic severe accident analysis, the code suite had some significant short-comings. Since the code represented the linkage of many separate code modules, the data transfer and feedback effects were not always handled consistently. The technical basis for the models in the STCP was documented in NUREG-0956 [14]. The results from the STCP calculations supported the NUREG-1150 probabilistic risk assessment [5] along with expert judgment and simplified algorithms for sequence-specific source terms.

The NUREG-1150 probabilistic (but not in consequence calculations) risk assessment was source term an effort to put the insights gained from the research on system behavior and phenomenological aspects of severe accidents into a risk perspective. An important characteristic of this study was the inclusion of the uncertainties in the calculations of core damage frequency and ~~risk~~ due to incomplete understanding of reactor systems and severe accident phenomena. The elicitation of expert judgment was used to develop probability distributions for many accident progression, containment loading, structural response, and source term issues. As noted in NUREG-1150, "computer analyses cannot, in general, be used directly and alone to calculate branching probabilities in the accident progression event tree. Since the greatest source of uncertainty is typically associated with the modeling of severe accident phenomena, the results of a single computer run (which uses a specific model) do not characterize the branching uncertainty." It was therefore necessary to use sensitivity studies, uncertainty studies, and expert judgment to characterize the likelihood of alternative events that affect the course of an accident. ~~The elicitation of expert judgment was used to develop probability distributions for many accident progression, containment loading, and structural response issues.~~ The insights from the NUREG-1150 study have been used in several areas of reactor regulation including the development of alternative radiological-source terms for evaluating design basis accidents at nuclear reactors.

See this earlier in the same paragraph!

In 1995, the NRC published NUREG-1465 [13], which defined an alternative accident source term for regulatory applications. The NUREG-1465 source term is considered an alternative to TID-14844, which specified a release of fission products from the core to the reactor containment in the event of a postulated accident involving a "substantial meltdown of the core." NUREG-1465 documents the basis for more realistic estimates of the source term release into containment, in terms of timing, nuclide types, quantities, and chemical form, given a severe core-melt accident. This revised source term is to be applied to the design of future light water reactors (LWRs). Current LWR licensees may voluntarily propose applications based upon it.

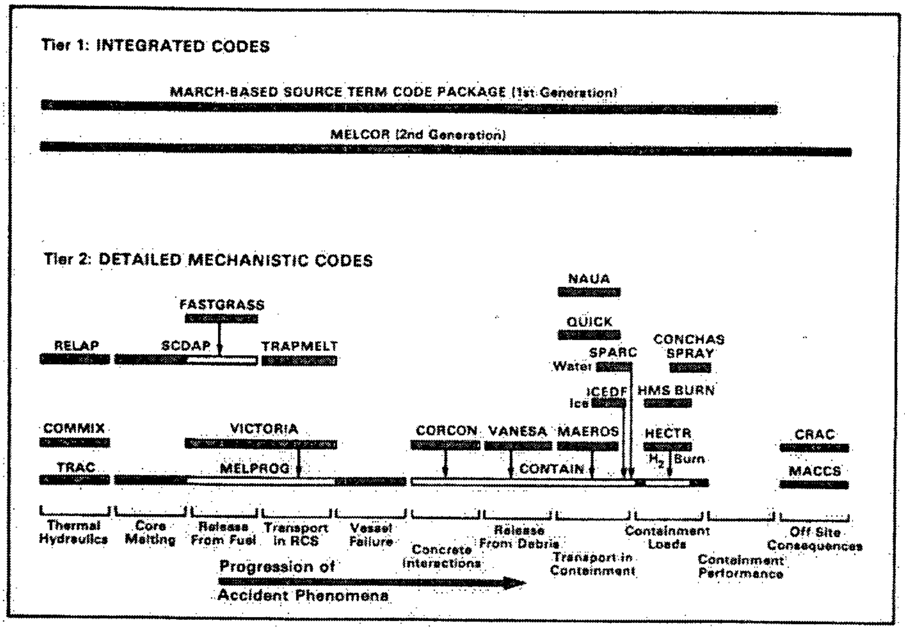


Figure 3 Summary of NRC Codes and Two Tier Structure (circa 1986) [14]

4.2 NRC Severe Accident Codes

As a consequence of the need to perform calculations covering a broad range of phenomena, a two-tier code strategy was developed by NRC in the 1980s (see Figure 3). The STCP was the first Tier 1 integrated analysis code. It was capable of calculating the full scope of the severe accident progression including the radionuclide source term. The STCP was a coupling of ten separate codes that were independently developed to calculate specific aspects of the severe accident progression (e.g., the CORSOR code predicted in-vessel fission product releases and the CORCON code evaluated ex-vessel core-concrete interactions). The Tier 1 codes were originally conceived to include modeling simplifications in order to permit calculation of all phases of the accident. In response to problems associated with coupling many different codes, the MELCOR code development program was initiated to develop a fully integrated code with flexible nodalization capabilities, intrinsic and self-consistent feedback between phenomena, and sensitivity analysis capabilities.

The second code tier of severe accident codes that were developed by the NRC was called the detailed mechanistic codes. The detailed mechanistic codes were typically developed and applied in close connection with an experimental program. Their scope was often limited to planning and interpreting experiments. However, the level of detail often far exceeded the comparable models in the Tier 1 codes. Therefore, the mechanistic codes, or the scientific principles within them, are subsequently used to enhance the integrated codes (i.e., MELCOR). In short, the science of severe accident phenomena is developed in the mechanistic codes and transferred to the integrated codes [14].

Over time, the number of codes maintained by the NRC decreased. MELCOR became the only Tier 1 integrated code and SCDAP/RELAP5, VICTORIA, and CONTAIN were the primary Tier 2 mechanistic codes. SCDAP/RELAP5 calculated the thermal-hydraulic and severe accident response of the primary and secondary systems of the nuclear reactor. It was not used for the radionuclide release and transport calculations. VICTORIA is a detailed code for prediction of the chemical forms of fission products in the primary reactor coolant system. However, it requires thermal-hydraulic boundary conditions from a primary system analysis code. CONTAIN only calculates the containment response and ex-vessel severe accident phenomena.

The MELCOR code was originally conceived as a Tier 1 integrated analysis code for probabilistic risk assessments. In particular, MELCOR models the full-scope of a severe accident including the source term but in a less detailed manner than the detailed mechanistic codes. However, the level of modeling detail in the MELCOR code steadily increased in the 1990s as computer processor speeds increased. The functionality of most of the detailed mechanistic separate effects codes in Figure 3 were fully integrated into MELCOR (e.g., VANESA, CORCON, and SPARC92). Starting in 2000, the NRC initiated a final code consolidation effort to incorporate the SCDAP/RELAP5, VICTORIA, and CONTAIN codes into MELCOR. Once complete, this will provide an increase in efficiency by requiring the maintenance of only one fully integrated code for severe accident analysis (see Figure 4). The scope of the MELCOR code is further discussed in Section 4.3.

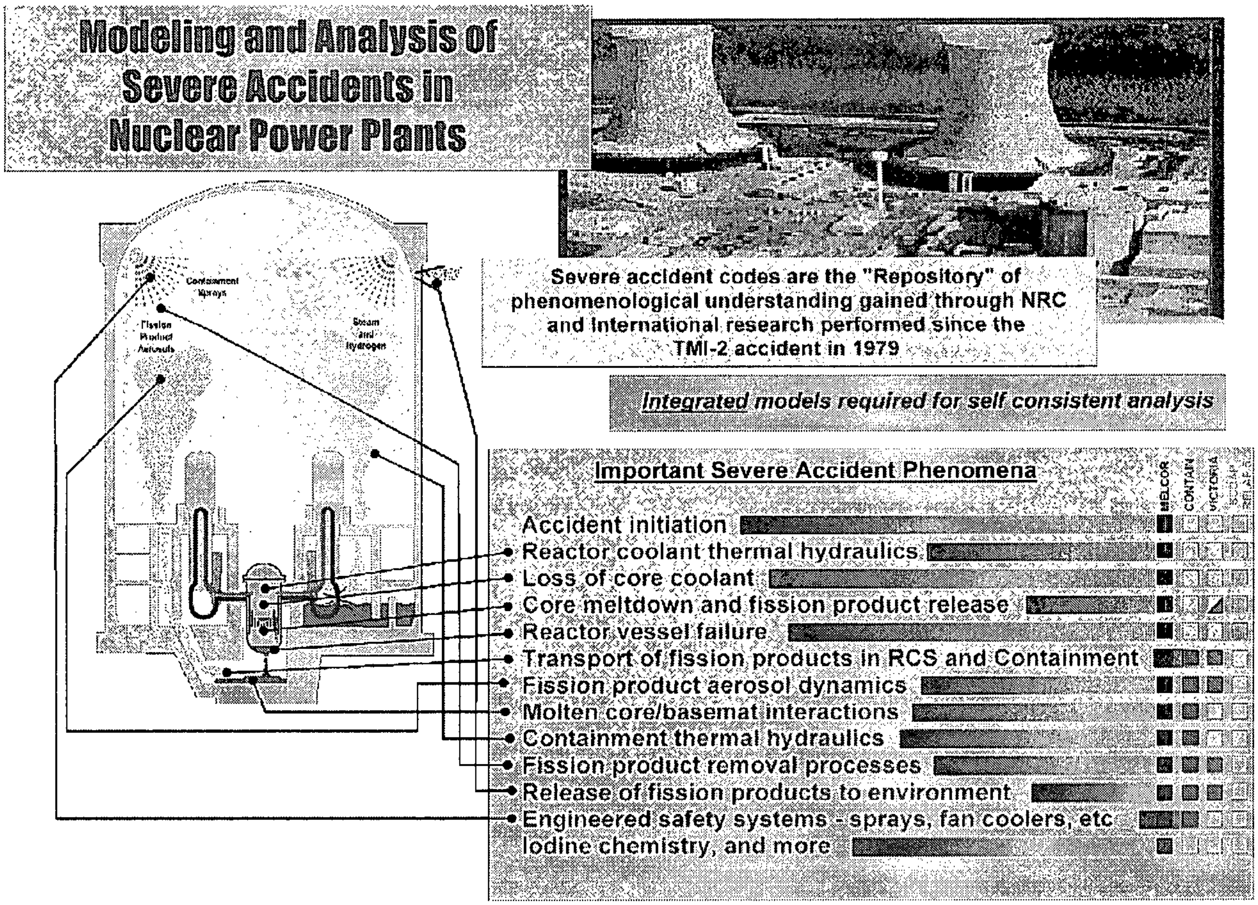


Figure 4 MELCOR Integration of Separate Effects Codes.

4.3 The MELCOR Code

The MELCOR code is a fully integrated, engineering-level computer code whose primary purpose is to model the progression of accidents in light water reactor nuclear power plants as well as in non-reactor systems (e.g., spent fuel pool, dry cask). Current uses of MELCOR include estimation of fission product source terms and their sensitivities and uncertainties in a variety of applications. MELCOR is a modular code comprised of three general types of packages: (a) basic physical phenomena (i.e., hydrodynamics (control volume and flow paths), heat and mass transfer to structures, gas combustion, aerosol and vapor physics); (b) reactor-specific phenomena (i.e., decay heat generation, core degradation, ex-vessel phenomena, sprays and engineering safety systems); (c) support functions (thermodynamics, equations of state, other material properties, data handling utilities, equation solvers). As a fully integrated code, MELCOR models all major systems of a reactor plant and their important coupled interactions.

The scope of MELCOR includes:

- thermal-hydraulic response of the primary reactor coolant system, reactor cavity, containment, and confinement buildings,
- core uncovery (loss of coolant), fuel heatup, cladding oxidation, fuel degradation (loss of rod geometry), and core material melting and relocation,
- heatup of reactor vessel lower head from relocated core materials and the thermal and mechanical loading and failure of the vessel lower head, and transfer of core materials to the reactor vessel cavity,
- core-concrete attack and ensuing aerosol generation,
- in-vessel and ex-vessel hydrogen production, transport, and combustion,
- fission product release (aerosol and vapor), transport, and deposition
- behavior of radioactive aerosols in the reactor containment building, including scrubbing in water pools, and aerosol mechanics in the containment atmosphere such as particle agglomeration and gravitational settling, and
- the impact of engineered safety features on thermal-hydraulic and radionuclide behavior.

Most MELCOR models are mechanistic and the use of parametric models ~~are~~ is limited to areas of high phenomenological uncertainty where there is no consensus concerning an acceptable mechanistic approach. Current use of MELCOR often includes uncertainty analyses and sensitivity studies. To facilitate this, many of the mechanistic models have been coded with optional adjustable parameters. This does not affect the mechanistic nature of the modeling, but it does allow the analyst to easily address questions of how particular modeling parameters affect the course of a calculated transient. Parameters of this type, as well as such numerical parameters as convergence criteria and iteration limits, are coded in MELCOR as sensitivity coefficients, which may be modified through optional code input. It should be noted that core radioactive nuclide inventories are not utilized by MELCOR, rather masses and decay heats of chemical element groups are used. Appropriate auxiliary code calculations are performed for specific fuel and core design and are carried out to the burnup of interest in order to provide the initial core inventories for MELCOR severe accident analysis (see Section 4.4.1).

4.4 MELCOR Modeling Approach

A high-level description of the MELCOR models that were used for the SOARCA project is presented in Section 4.4.1. Existing MELCOR models for Surry and Peach Bottom were updated to current state-of-the-art modeling practices ~~as well as the latest version of the MELCOR code.~~ More detailed information describing the plant models is provided in the plant-specific analysis reports (i.e., Appendix A and Appendix B for Peach Bottom and Surry, respectively).

[illegible] more detail next page

The progression of events in a severe accident contains uncertainties. The procedure to define the best practices approach to modeling important and uncertain phenomena is described in Section 4.4.2. NUREG/CR-7008 [61] provides a more detailed description of the best practices modeling approach.

Section 4.4.3 summarizes some recent changes to the radionuclide release and cesium speciation modeling, which is important to the source term results. Finally, the methodology to calculate the radionuclide inventory is described in Section 4.4.4.

4.4.1 Plant Models

The MELCOR models used in the SOARCA source term calculations represented the state-of-the-art. As part of the SOARCA program, the MELCOR models were updated to the most recent version of the MELCOR code.[2] The scope of the models included:

- Detailed 5-ring reactor vessel models
- Representation of the primary reactor coolant systems (and secondary steam generator through the main steam isolation valve for Surry)
- Representation of the primary containment release
- Representation of the Peach ~~Bottom~~ reactor building and the Surry auxiliary building, which were radionuclide pathways in some scenarios
- Representation of the emergency core-cooling systems (and the auxiliary feedwater system for Surry)
- Representations of the emergency, portable water injection systems

Through the best practices updates to each deck, the following new models were specified for both plants for these important but uncertain phenomena or equipment responses.

- Safety relief valve failure models for normal or high temperature conditions,
- An additional thermo-mechanical fuel collapse model for heavily oxidized fuel following molten zircaloy breakout,
- Enhanced lower plenum coolant debris heat transfer that recognizes break-up and multi-dimensional cooling effects not present in the one-dimensional counter-current flooding model in older versions of MELCOR (e.g., [23]),
- Updated, plant-specific chemical element masses and decay heats (see Section 4.4.4),
- A new ORNL-Booth chemical element release model and new Cs speciation model (see Section 4.4.3),
- Vessel failure based on gross failure[3] [24] using the improved one-dimensional creep rupture model with the new hemispherical head model and radial heat transfer between lower head conduction node segments, and
- Enhanced ex-vessel core debris heat transfer that recognizes multi-dimensional effects and rates measured in MACE tests [25].

ordinary all SOARCA calculations / If this is true, don't bring up V2.0! But I think V2.0 was actually used

[2] ~~MELCOR Version 1.8.6 was used for all SOARCA calculations.~~ MELCOR Version 2.0 was released during the initial phase of the SOARCA program. Version 2.0 is based on identical physics models as Version 1.8.6 but has been modernized to use FORTRAN 90, new input format, and to enable automated source term information for preparing MACCS2 input. to use

[3] A more complete discussion of this model is presented in NUREG/CR-7008 [61] and the MELCOR manual [15]. A penetration failure model was not used, because the timing differences between gross lower head failure and penetration failure with the available penetration model is not significant to the overall accident progression (i.e., minutes difference). Also, Sandia Lower Head Failure (LHF) tests showed gross creep rupture of the lower head was ~~measured to be~~ the most likely mechanism for vessel failure [24].

A summary of recent enhancements to the MELCOR Peach Bottom and Surry models for the SOARCA program are presented in Section 4.4.1.1 and 4.4.1.2, respectively.

4.4.1.1 Peach Bottom MELCOR Model

The Peach Bottom MELCOR model was originally developed for code version 1.8.0 at Brookhaven National Laboratories. The model was subsequently adopted by J. Carbajo at Oak Ridge National Laboratories to study differences in fission product source term behavior predicted by MELCOR 1.8.1 and those generated for use in NUREG-1150 using the Source Term Code Package (STCP) [26]. Starting in 2001, Sandia National Laboratories made considerable refinements to the BWR/4 core nodalization to support the developmental assessment and release of MELCOR 1.8.5. These refinements concentrated on the spatial nodalization of the reactor core (both in terms of fuel/structural material and hydrodynamic volumes) used to calculate in-vessel melt progression.

Subsequent work in support of several ~~U.S.~~ NRC research programs has motivated further refinement and expansion of the BWR/4 model in four broad areas. The first area involved the addition of models to represent a wide spectrum of plant design features, such as safety systems, to broaden the capabilities of MELCOR simulations to a wider range of severe accident sequences. These enhancements include:

- modifications of modeling features needed to achieve steady-state reactor conditions (recirculation loops, jet pumps, steam separators, steam dryers, feedwater flow, control rod drive hydraulic system, main steam lines, turbine/hotwell, core power profile),
- new models and control logic to represent coolant injection systems (RCIC, HPCI, RHR, LPCS) and supporting water resources (e.g., CST with switchover), and
- new models to simulate reactor vessel pressure management (safety relief valves, safety valves, automatic depressurization system, and logic for manual actions to effect a controlled depressurization if torus water temperatures exceed the heat capacity temperature limit).

The second area focused on the spatial representation of primary and secondary containment. The drywell portion of primary containment has been sub-divided to distinguish thermodynamic conditions internal to the pedestal from those within the drywell itself. Also, refinements have been added to the spatial representation and flow paths within the reactor building (i.e., secondary containment). The third area has focused on bringing the model up to current "best practice" standards for MELCOR 1.8.6 (see Section 4.4.2). The fourth area of model improvements included a new radionuclide inventory and decay heat based on the recent plant operating history (see Section 4.4.4).

While not new for SOARCA, the MELCOR Peach Bottom model includes a multi-region ex-vessel debris spreading model. The debris spreads according to its temperature relative to the solidus and liquidus temperatures of the concrete and the debris height. If the debris spread against the drywell liner steel wall, the liner will fail if the debris temperature is above the carbon steel melting temperature.

The MELCOR Peach Bottom model is more fully described in Appendix A. The MELCOR nodalization diagrams for Peach Bottom are shown in Figure 5.

4.4.1.2 Surry MELCOR Model

The Surry MELCOR model applied in this syudy was originally generated at Idaho National Engineering Laboratories (INEL) in 1988. The model was periodically updated by Sandia National Laboratories (1990 to present) for the purposes of testing new models, advancing the state-of-the-art in modeling of PWR accident progression, and providing support to decision-makers at the NRC for analyses of various issues that may affect operational safety. Significant changes were made during the last twenty years in the approach to modeling core behavior and core melt progression, as well as the nodalization and treatment of coolant flow within the RCS and reactor vessel. In 2002, the reactor vessel and RCS nodalization were updated using the SCDAP/RELAP5 Surry model to include a five ring vessel nodalization and counter-current hot leg representation for natural circulation flow [27]. The current MELCOR Surry model is a culmination of these efforts and represents the state-of-the-art in modeling of potential PWR severe accidents.

In preparation for the SOARCA analyses described in this report, the model was further refined and expanded in three areas. The first area ~~is~~ was an upgrade to MELCOR Version 1.8.6 core modeling. These enhancements include~~:~~d

- a hemispherical lower head model that replaces the flat bottom-cylindrical lower head model,
- new models for the core former and shroud structures that are fully integrated into the material degradation modeling, including separate modeling of debris in the bypass region between the core barrel and the core shroud,
- models for simulating the formation of molten pools in both the core and lower plenum, crust formation, convection in molten pools, stratification of molten pools into metallic and oxid~~e~~ic layers, and partitioning of radionuclides between stratified molten pools,
- a reflood quench model that separately tracks the component quench front, and the quenched and unquenched temperatures,
- a control rod silver aerosol release model, and
- addition of the new ONRL-Booth radionuclide release model for modern high-burn-up fuel.

The second area focused on the addition of user-specified models to represent a wide spectrum of plant design features and safety systems to broaden the capabilities of MELCOR to a wider range of severe accident sequences. These enhancements included:

- update of the containment leakage model,
- update of core degradation modeling practices,
- modeling of individual primary and secondary relief valves with failure logic for rated and degraded conditions,
- update of the containment flooding characteristics,
- heat loss from the reactor to the containment,

- separate motor and turbine-driven auxiliary feedwater models with control logic for plant automatic and operator cooldown responses,
- new turbine-driven auxiliary feedwater models for steam flow, flooding failure, and performance degradation at low pressure,
- nitrogen discharge model for accumulators,
- update of the fission product inventory, the axial and radial peaking factors, and an extensive fission product tracking control system, and
- improvements to the natural circulation in the hot leg and steam generator and the potential for creep rupture.

The MELCOR Surry model is more fully described in Appendix B. The MELCOR nodalization diagrams for Surry are shown in Figure 6.

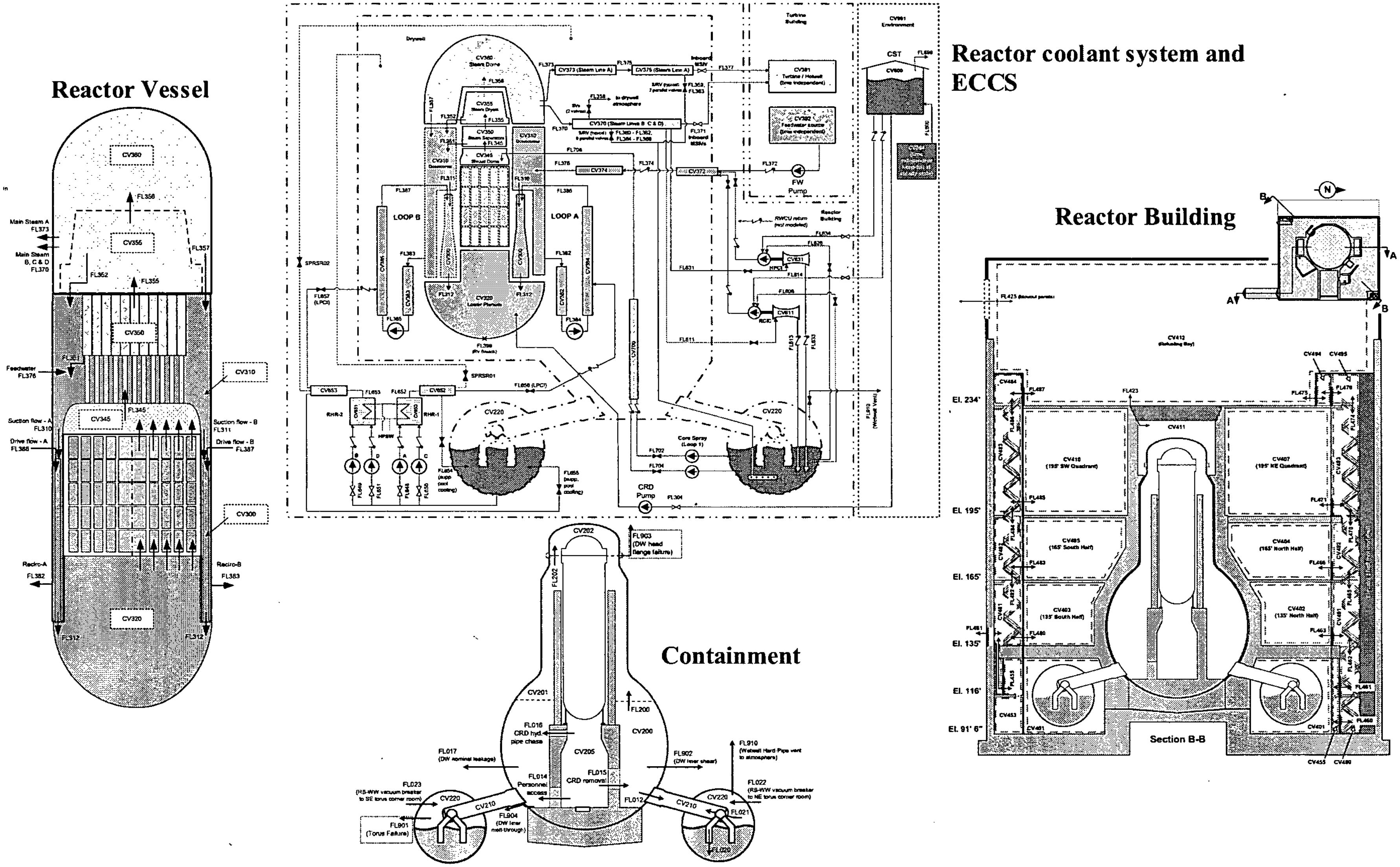


Figure 5 The Peach Bottom MELCOR Nodalization.

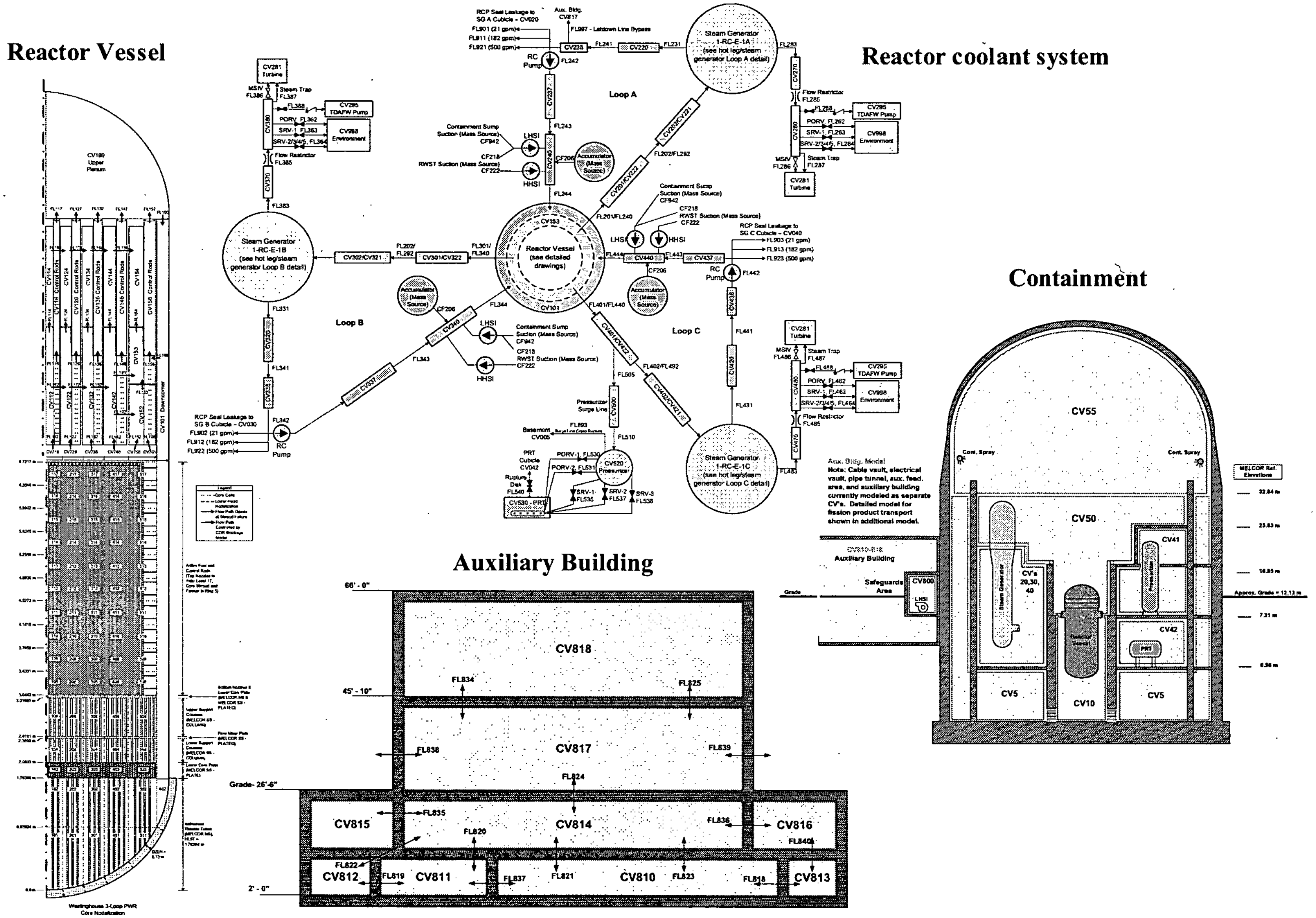


Figure 6 The Surry MELCOR Nodalization.

4.4.2 Best Modeling Practices

The accident progression analysts developed a list of key uncertain phenomena that can have a significant effect on the progression of the accident. Each issue was outlined and a recommended modeling approach or base case values were identified in plant-specific reports for Peach Bottom (Appendix A) and Surry (Appendix B). A discussion of the specific modeling practices are described in NUREG/CR-7008 [61].

For operator actions, a sensitivity calculation was performed for each accident sequence to quantify the impact of the operator response.

Is this an oblique reference to the mitigated vs unmitigated calcs? If so, this only since its already mentioned a sufficient number of times. If not, expand a bit so the reader has some idea of what was really investigated

Several early containment failure modes of historical interest were excluded from the SOARCA project due their assessed low-likelihood of occurance. These include:

1. *Alpha mode containment failure*, which is an in-vessel steam explosion during melt relocation that simultaneously fails the vessel and the containment. A group of leading experts in this field, referred to as the Steam Explosion Review Group, concluded in a position paper published by the Nuclear Energy Agency Committee on the Safety of Nuclear Installations [28] that the alpha-mode failure issue for Western-style reactor containment buildings can be considered resolved from a risk perspective, having little or no significance to the overall risk from a nuclear power plant.

2. *Direct containment heating* (DCH) causing containment failure in PWR containments. Decades of NRC research show an early failure of the reactor coolant system due to high temperature natural circulation will depressurize the system prior to vessel failure. In the unlikely event there is a high-pressure vessel failure, the resolution of the DCH issue found ~~the~~ early containment failure to be very unlikely [29].

3. *Early containment failure due to drywell liner melt-through in a wet cavity in Mark I containments* (e.g., Peach Bottom). Through a detailed assessment of the issue, it was concluded that, in the presence of water, the probability of early containment failure by melt-attack of the liner is so low as to be considered physically unreasonable [30].

An independent expert panel was assembled to review the proposed modeling approach for SOARCA analyses. This review was conducted during a public meeting sponsored by the NRC on August 21–22, 2006 in Albuquerque, New Mexico. The expert panel review examined the best modeling practices for the application of the severe nuclear reactor accident analysis code MELCOR for realistic evaluation of accident progression and source term ~~and offsite consequences~~. The panel also reviewed a set of code enhancements as well as consideration of the SOARCA project in general.

[60] ?

4.4.3 ~~Radionuclide~~ Chemical element Modeling

as discussed below

The ~~radionuclide~~ chemical element modeling was updated in the Peach Bottom and Surry models to apply a more mechanistic ~~radionuclide~~ release model (i.e., the ORNL-Booth model) [31] based on assessments ~~to~~ of recent radionuclide release tests. These assessments identified an alternative set of Booth diffusion parameters recommended by ORNL (ORNL-Booth) [32], which produced significantly improved release signatures for Cs and other fission product groups. Some adjustments to the scaling factors in the ORNL-Booth model were made for selected fission product groups, as including UO_2, Mo and Ru, in order to gain better comparisons with the FPT-1 ~~data~~ [33]. The adjusted model, referred to as "Modified ORNL-Booth," was subsequently compared to original ORNL VI fission product release experiments and to more recently performed French VERCORS tests [34], and the comparisons ~~was~~ as favorable or better than the original CORSOR-M MELCOR default release model. These modified ORNL-Booth parameters were implemented into the MELCOR code as new defaults for the SOARCA project.

insert

While significant improvements in release behavior were obtained for the analysis of the FPT-1 test with the ORNL-Booth parameters, some additional modification to the MELCOR release model was pursued. Evidence from the Phebus experiments increasingly indicates that the dominant chemical form of released Cs is that of Cs_2MoO_4. This is based on deposition patterns in the Phebus experiment where Cs is judged to be in aerosol form at 700C, explaining deposits in the hot upper plenum of the Phebus test section, and deposition patterns in the cooler steam generator tubes. In recognition of ~~response~~, a Cs_2MoO_4 radionuclide class was defined with the vapor pressure Cs_2MoO_4 and the release coefficients developed for Cs. While having little effect on the net release of Cs, this change had a significant effect on the release of Mo. The Mo vapor pressure is so exceedingly low that the net release is limited by the vapor pressure transport term. Since there is significantly more Mo than Cs in the radionuclide inventory, only a portion of the Mo was added to the new Cs_2MoO_4 ~~radionuclide~~ chemical element class. this experimental result

Chemical element

The ~~radionuclide~~ chemical element input was reconfigured to (a) represent the dominant form of ~~Cs~~ chemical element as Cs_2MoO_4, (b) represent the dominant form of I as CsI, and (c) represent the gap inventories consistent with the NUREG-1465 recommendations [13]. The MELCOR ~~radionuclide~~ transport, deposition, condensation/evaporation, and scrubbing models were all activated. The model for chemisorption of Cs to stainless steel was activated. In addition, the hygroscopic coupling between the steam/fog condensation/evaporation thermal-hydraulic solutions ~~to~~ and the airborne aerosol size and mass was also activated [31].

4.4.4 ~~Radionuclide~~ Chemical Element Inventory

One important input to MELCOR is the initial mass of the ~~radionuclides~~ chemical elements in the fuel and their associated decay heat [31]. These values are important to the timing of initial core damage and the location and concentration of the radionuclides in the fuel. The radio-isotopes in a nuclear reactor come from three primary sources: (1) "fission products" are the result of fissions in either fissile or fissionable material in the reactor core, (2) actinides are the product of neutron capture in the initial heavy metal isotopes in the fuel, and (3) other radio-isotopes are formed from the radioactive decay of these fission products and actinides. Integrated computer models such as the TRITON sequence in SCALE exist to capture all of these inter-related physical processes, but they are intended primarily as reactor physics tools [35]. As such, their standard output does not provide the type of information needed for SOARCA. Therefore, a method of deriving the needed information is described in this report. It is important to note upfront that no changes to the physics codes were needed. The method described here merely extracts additional output from the TRITON sequence and combines it in a way which makes it useful for the SOARCA project.

4.4.4.1 Description of Relevant Physics

The concentration of ~~radio-isotopes~~ Nuclides at any time in the reactor is governed by the relationship shown in Equation 1 (note that not all radio-isotopes are born from fission or lost by capture or radioactive decay and that these concentrations are generally spatially dependent).

$$\frac{dC_i}{dt} = \sum_n \gamma_n \Sigma_f(t)\theta(t) - \Sigma_a(t)\theta(t) - \lambda N(t) + \sum_n \lambda_n N_n(t) \qquad (1)$$

The right hand side of this equation contains four terms capturing, respectively, the following effects: (1) creation from fission; (2) loss due to absorption; (3) loss due to radioactive decay; and (4) creation due to radioactive decay of parent nuclides. All of these processes are time dependent and are either directly or indirectly linked to the magnitude of the neutron flux. Therefore, in order to correctly predict the radio-isotopic concentration in a reactor core one must account for both the reactor burnup and the specific power.

4.4.4.2 Methods

Reactor physics codes implicitly account for both of the physical parameters of interest for SOARCA (i.e., decay heat power and ~~radionuclide~~ chemical element inventories), but they do not provide a mechanism to easily extract and combine these results. This section will describe the tools used to calculate the ~~radio-isotopic~~ inventory and a new code developed to properly combine these results for use in the SOARCA calculations. The results were combined in a manner so as to capture actual plant operating data.

The TRITON sequence from SCALE 5.1 was used to develop input data for MELCOR. TRITON provides the capability to perform detailed two-dimensional calculations of reactor fuel including the ability to deplete fuel to a user defined level of accuracy. TRITON accurately models curvilinear surfaces such as cylindrical fuel rods and allows the fuel to be burned down to the sub pin-cell level. There is no requirement to perform any homogenization of the two-dimensional geometry. TRITON allows for accurate depletion of highly self shielded fuel such as poison pins. For more information, refer to the SCALE documentation [36].

The BLEND3 code was developed from previous work performed by Oak Ridge National Laboratory and its capabilities were extended for this study. BLEND3 uses the reactor-specific fuel loading from three different cycles, the nodal exposure, and the assembly specific power data from the licensee to derive node averaged radio-isotopic inventories. TRITON uses a generic fuel assembly data and ties it to specific reactor operating conditions. Then, BLEND3 performs the following tasks. First, for a given node, BLEND3 identifies which specific power ORIGEN output files are assigned to the specified input power. Second, for three different cycles of fuel, BLEND3 interpolates a radio-isotopic inventory from the relevant ORIGEN output files. Finally, using the input volume fractions for the three different cycles of fuel, it creates a new, volumetrically averaged ORIGEN output file for the node for the specified input conditions.

The PRISM module from SCALE 5.1 was then used to drive ORIGEN decay calculations using the newly created averaged ORIGEN output files as input. PRISM is a SCALE utility module which allows the user to automate the execution of a series of SCALE calculations.

4.4.4.3 Peach Bottom Model

The Peach Bottom model is based on the Global Nuclear Fuel (GNF) 10x10 (GE-14C) fuel assembly. The GNF 10x10 is representative of a limiting fuel type actually being used in commercial BWRs. The GEH 10x10 model is illustrated in Figure 7. The model is very detailed for this application. The only significant assumption was that the part length rod portion of the reactor was modeled as a full assembly.

Twenty-seven different TRITON runs were performed to model three different cycles of fuel at nine different specific power histories. The specific power histories ranged from 2 MW/MTU to 45 MW/MTU to cover all expected BWR operational conditions. For times before the cycle of interest, an average specific power of 25.5 MW/MTU was used. For example, for second cycle fuel, the fuel was burned for its first cycle using 25.5 MW/MTU, allowed to decay for an assumed 30 day refueling outage and then 9 different TRITON calculations were performed with specific powers ranging from 2 to 45 MW/MTU. The BLEND3 code was then applied to each of the 50 nodes in the MELCOR model using the average specific powers and volume fractions. Once new libraries for each of the 50 nodes in the model were generated, the final step in the procedure was to deplete each node for 48 hours. The decay heats, masses, and specific activities as a function of time were processed and applied as input data to MELCOR to define decay heat and the ~~radionucl~~ide inventory. Also, the ___ values were reanalyzed to provide the inventories of the radionuclides used by MACCS2 at shutdown. & the calculated

chemical element

core-wide

The depletion for the figure of interest was to the mid-cycle point.

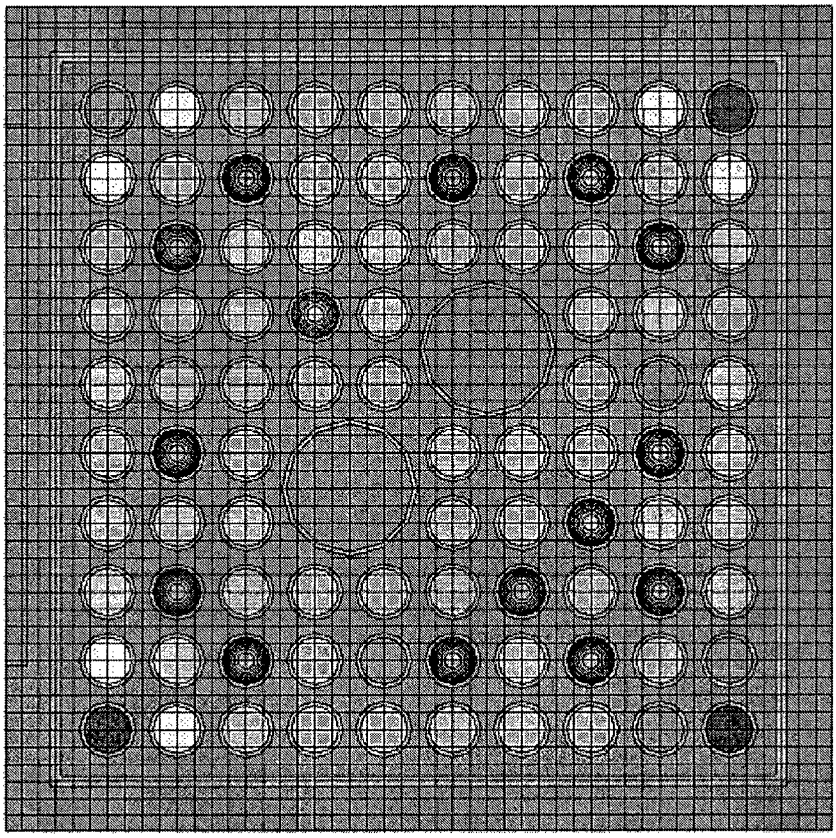


Figure 7 Schematic of Modeling Detail for BWR GNF 10x10 Assembly.

4.4.4.4 Surry Model

Previously, detailed input was developed for Surry in a separate NRC program on the source term from high-burnup uranium (HBU) fuel at the end of the fuel cycle. This previous study used the same methodology as the Peach Bottom model (Section 4.4.4.3). The actual mid-cycle decay power is lower. However, ~~due to schedule constraints,~~ the Surry model did not include a current operation, decay heat evaluation as was done for Peach Bottom.

hardly a reason after all these years!

4.4.4.5 Evaluation of the Results

There are very few measurements of decay heat in existence and those that do exist are not directly relevant to this study. Therefore, the discussion of the decay heat predictions will be limited to a comparison to previously published work. The best known source of decay heat predictions is summarized in Regulatory Guide 3.54 and results from the guide will be used to assess the predictions in the current study [37]. Decay heat for two decay times will be used as a check on the consistency of the results presented in this study. By interpolation of tables in RG 3.54 for a specific power of 27 MW/MTU, decay powers at 1 and 2 years following shutdown of 9.3 W/kgU and 5.1 W/kgU, respectively, are calculated. Using the results from the Peach Bottom calculations, the corresponding decay powers are 8.92 W/kgU and 4.734 W/kgU. The maximum difference between results is approximately 8 percent which is considered acceptable given the best estimate nature of the SOARCA study compared to the methods used to generate the tables in RG 3.54.

A quantitative discussion of the radio-isotopic predictions presented in this study would be of limited use given the cycle specific nature of this work. However, it is of benefit to discuss the relevant SCALE assessment. Specifically, the TRITON module has been assessed by M. D. DeHart and S. M. Bowman [38], S. M. Bowman and D. F. Gill [39], and Germina Ilas and Ian C. Gauld [40]. These assessment reports use data from Calvert Cliffs, Obrigheim, San Onofre, and Trino Vercelles PWRs. The third report summarized comparisons to decay heat measurements from 4 different BWR assemblies.

We need to add a section on containment behavior — hydrogen deflagration/detonation, α, DCH, wet liner melt through — leak behavior vs large hole — but it Does not have to be long, is an important aspect and should be cut + pasted into this MELCOR area — It's briefly mentioned in the results section (pg 66) but deserves more.

For P.B. need to talk about liner melt through delayed ~~compared~~ compared to other studies because of core melt delay processes

6.0 RESULTS AND CONCLUSIONS

To assess the benefits of the various mitigative measures (which were scenario specific) ~~and to provide a basis for comparison to past analyses of unmitigated severe accident scenario~~s, the SOARCA project analyzed the selected scenarios assuming 10CFR50.54(hh) procedures were successful (mitigated) as well assuming they were unsuccessful (unmitigated). The following sections summarize the results of the Peach Bottom and Surry accident progression and offsite consequence results. Greater detail regarding these results is provided in the appendices to this NUREG.

6.1 Accident Progression and Radionuclide Release

The assessment of mitigation measures, including emergency operating procedures, severe accident management guidelines, and security-related mitigation measures, led the staff to conclude that all the identified scenarios could reasonably be mitigated. The security-related measures to provide alternative AC power and portable diesel-driven pumps were especially helpful in counteracting SBO scenarios. For the ISLOCA scenario, installed equipment ~~was~~ that adequate to prevent core damage owing to the time available for corrective ~~action~~. For all events except one, the mitigation was sufficient to prevent core damage. For one event, the Surry short-term SBO, the mitigation was sufficient to enable flooding of the containment through the containment spray system to cover core debris. The assessment of the mitigation measures was undertaken with support from integrated accident progression analyses using the MELCOR code. MELCOR analyses were used to both confirm the time available to take mitigation measures and to confirm that those measures, once taken, were adequate to prevent core damage or significantly reduce radiological releases. In other instances, MELCOR analyses using only installed equipment revealed that PRA success criteria were overly conservative, indicating core damage, where MELCOR analysis indicated no core damage. In addition, the release was assuming truncated at 48 hours (72 hours for the Surry long-term SBO) as a result of continually escalating mitigation actions, involving both onsite and offsite resources, including containment and reactor building flooding.

An important result of the MELCOR accident progression analyses was the insight that accident progression in severe accidents proceeds much more slowly than earlier treatments indicated. The reasons for this are principally twofold—(1) research and development of better phenomenological modeling has produced a much more protracted and delayed core degradation transient with substantial delays of reactor vessel failure and (2) all aspects of accident scenarios receive more realistic treatment, which includes more complete modeling of plant systems, and often yields delays in core damage and radiological release. ~~In general, bounding approaches have been used in past simplified treatments using qualitative logical models.~~ In SOARCA, where specific self-consistent scenarios are analyzed in an integral fashion using MELCOR, the result is that accident conditions or attributes that contribute to a more severe response in one area may produce an ameliorating effect in another area.

In the most likely accidents considered in SOARCA (assuming no 10CFR50.54(hh) mitigation)—the long-term SBO—core damage was delayed for 10 to 16 hours and reactor vessel failure was delayed for approximately 20 hours. Approximately 20 hours (BWR) or 45 hours (PWR) were available before the onset of offsite radiological release due to containment failure. ~~In the most widely referenced siting study scenario (identified as the SST1 release), it was assumed that a major release occurs in 1½ hours.~~ The SOARCA analyses showed that ample time is available for operators to take corrective action and for input from plant technical support centers even if initial efforts are assumed unsuccessful. Even in the case of the most rapid events (i.e., the unmitigated short-term SBO where core damage may begin in 1 to 3 hours), reactor vessel failure is delayed for roughly 8 hours allowing time for restoration of cooling and preventing vessel failure. In these cases, containment failure and radiological release is delayed for 8 hours (BWR) or 24 hours (PWR). For the bypass ~~events, substantial~~ delays occur or, in the case of the thermally induced steam generator ~~tube~~ rupture, the radiological release is shown by analyses to be ~~substantially reduced~~ small. Table 4 for Peach Bottom and Table 5 for Surry provide key accident progression timing results for SOARCA scenarios. Table 4 shows the same times for lower head failure and start of the release to the environment, because drywell shell melt-through occurs about 15 minutes after lower head failure. ~~(co...)~~

Table 4 Peach Bottom Accident Progression Timing Results

Scenario	Time to start of core damage (hours)	Time to lower head failure (hours)	Time to start of release to environment (hours)
Long-term SBO	10	20	20
Short-term SBO	1	8	8

Table 5 Surry Accident Progression Timing Results

Scenario	Time to start of core damage (hours)	Time to lower head failure (hours)	Time to start of release to environment (hours)
Long-term SBO	16	21	45
Short-term SBO	3	7	25
Thermally induced steam generator tube rupture	3	7.5	3.5
Interfacing systems LOCA	9	15	10

The SOARCA study also demonstrated that the magnitude of the radionuclide release is likely to be much smaller than ~~used~~ estimated in past studies, again as a result of extensive research and improved modeling and as a result of integrated and more complete plant simulation. Some releases of important radionuclides such as iodine and cesium are predicted to be about 10 percent, but are more generally in the range of 0.5 to 2 percent. By contrast, the siting study used an iodine release of 45 percent and a cesium release of 67 percent. Figure 12 and Figure 13 provide the radionuclide release results for iodine and cesium.

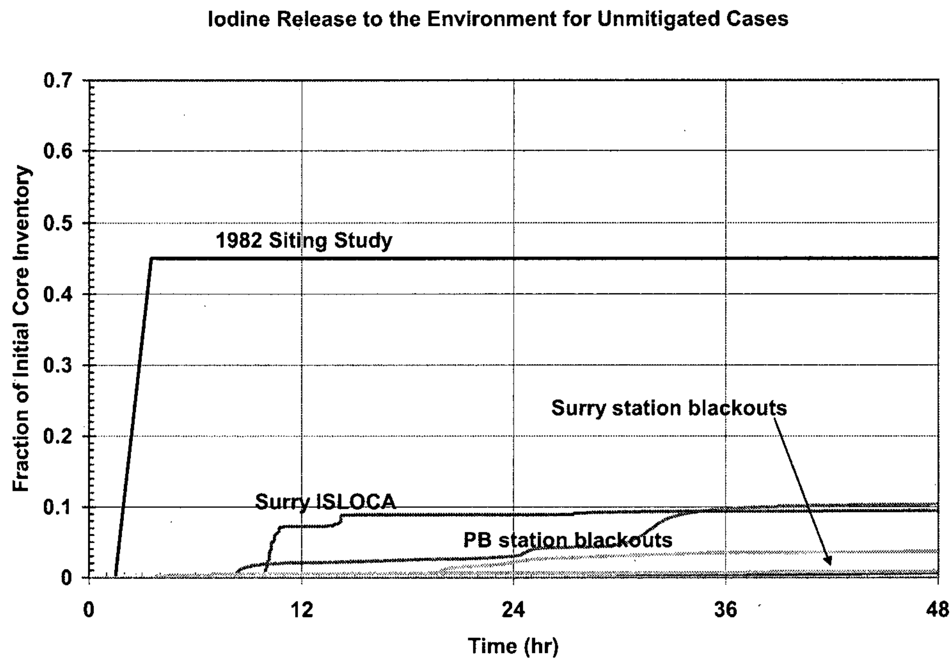
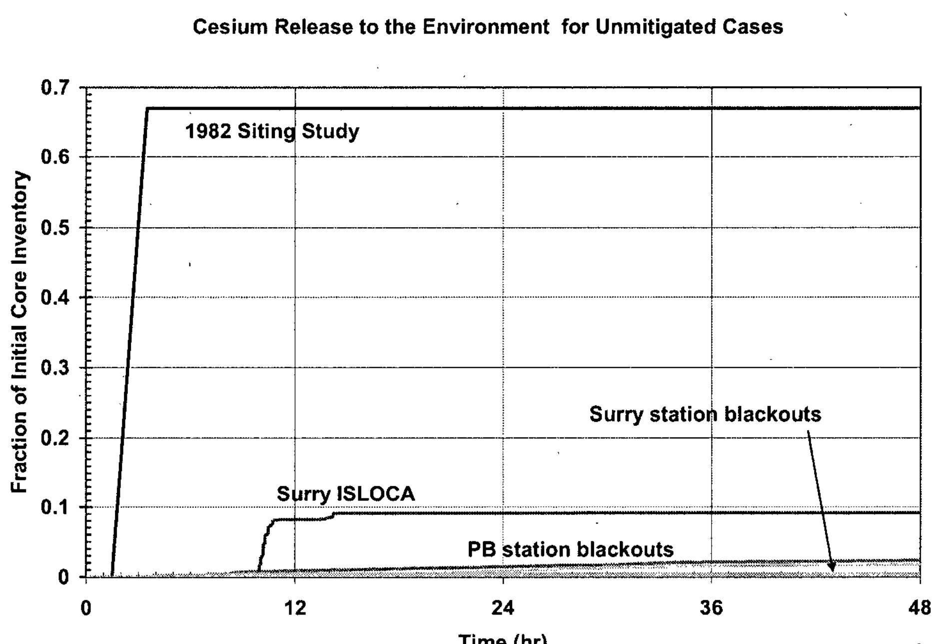


Figure 12. Iodine Releases to the Environment for SOARCA Unmitigated Scenarios



I would prefer not to compare CST1 with these results - In actuality, the SOARCA results are not comparable compared w/ CSSS because of the difference in containment behavior, in large increase. The results have not changed much - except LCF would be 3X larger - if comparable containment ~~beh~~ behavior source terms were compared -

Figure 13. Cesium Releases to the Environment for SOARCA Unmitigated Scenarios

Sequences involving large early releases have influenced the results of past PRAs and consequence studies. For example, the siting study results were co~~ntrolled by~~ an internally initiated event with a large early release that was assigned a representative frequency of $1x10^{-5}$/year, based on knowledge available at the time. However, in the SOARCA study, no sequences with a frequency above $1x10^{-7}$/year resulted in a large early release, even considering external events and unsuccessful mitigation. This is a result of research conducted over the last 2 decades that has shown that phenomena earlier believed to lead to a large early release are of extremely low probability or physically unfeasible. This research was focused on phenomena that have been previously assumed to be prime contributors to severe accident risk, including direct containment heating and alpha mode failure.

The PWR SBO with a thermally induced steam generator tube rupture has in the past been believed to result in a large, relatively early release potentially leading to higher offsite consequences. However, MELCOR analysis performed for SOARCA showed that the release was small owing to thermally induced failures of other reactor coolant system components after the tube rupture. Also, the release was somewhat delayed; for the short-term SBO where no injection occurred at the start of the accident, the tube rupture and release began about 3.5 hours into the event. Further, core damage, tube rupture, and radiological release could be delayed for many hours if auxiliary feedwater were available even for a relatively short time period.

6.2 Offsite Radiological Consequences, compared with previous analyses.

The result of the accident progression and source-term analysis is that releases are delayed and of a diminished magnitude. Because of this and the realistic simulation of emergency response, essentially no early fatalities were predicted, as close-in populations were evacuated before or shortly after plume arrival. but at a cost either in contaminated or interdicted land or in decontamination cost or dose.

Latent health effects calculated using any of the dose-response models referenced in this study are small in comparison to the Safety Goal. Much of the latent cancer risk was in fact derived from the relatively small doses received by populations returning to their homes in accordance with emergency planning guidelines. Because much of the risk is due to the eventual return of the population, it is therefore controllable. For example, for the Peach Bottom long-term SBO, for individuals living within the EPZ, 99 percent of the latent cancer risk derives from the long-term dose received by the population returning to their homes and being exposed to small radiation doses. Similarly, about 70 percent of the latent cancer risk to individuals within 50 miles is from returning home. The percentage is larger for the EPZ, due to its evacuation prior to the start of the release. Here, the prediction of latent cancer risk, though very small, is strongly influenced by the relationship between low-dose health effects modeling and criteria for allowing return of populations.

Estimates of conditional (i.e., assuming the accident has occurred) individual latent cancer risk range from roughly 10^{-3} to 10^{-4}, using the LNT dose response model (other dose models result in lower or much lower conditional risk). If one also accounts for the probability of the severe accident itself, without successful mitigation (denoted as the absolute risk below), the risk to an individual for an important severe accident scenario is on the order of 10^{-9} to 10^{-10} per reactor for Peach Bottom for Surry

year. These risk estimates are a million times smaller than the U.S. average risk of a cancer fatality of $2x10^{-3}$ per year. Table 6 and Table 7 provide the risk estimates for individual SOARCA scenarios without successful mitigation using the LNT dose response model. The risk estimates are based on an assumed truncation of the release at 48 hours (72 hours for the Surry long-term SBO) as a result of continually escalating mitigation actions, including containment and reactor building flooding. The core damage frequencies shown in Table 6 and Table 7 assume the probability of 10 CFR 50.54(hh) mitigation is zero.

assuming

Table 6 Peach Bottom Results for Scenarios Without Successful Mitigation and Assuming LNT Dose Response Model

Scenario	**Core damage frequency (per reactor-year)**	**Conditional risk of latent cancer fatality for an individual located within 10 miles**	**Absolute risk of latent cancer fatality for an individual located within 10 miles (per reactor-year)**
Long-term SBO	$3x10^{-6}$	$2x10^{-4}$	$6x10^{-10}$
Short-term SBO	$3x10^{-7}$	$2x10^{-4}$	$7x10^{-11}$

Table 7 Surry Results for Scenarios Without Successful Mitigation and Assuming LNT Dose Response Model

Scenario	**Core damage frequency (per reactor-year)**	**Conditional risk of latent cancer fatality for an individual located within 10 miles**	**Absolute risk of latent cancer fatality for an individual located within 10 miles (per reactor-year)**
Long-term SBO	$2x10^{-5}$	$5x10^{-5}$	$7x10^{-10}$
Short-term SBO	$2x10^{-6}$	$9x10^{-5}$	$1x10^{-10}$
Thermally induced steam generator tube rupture	$4x10^{-7}$	$3x10^{-4}$	$1x10^{-10}$
Interfacing systems LOCA	$3x10^{-8}$	$8x10^{-4}$	$2x10^{-11}$

To provide additional information on the potential range of health consequences, the SOARCA project has developed latent cancer risk estimates assuming the LNT (endorsed by NCRP) and a for Surry

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range of truncation doses below which the cancer risk is not quantified. Dose truncation values used for SOARCA included 10 mrem/year (representing ICRP), 620 mrem/year (representative US background radiation) and 5 rem/year with a 10 rem lifetime cap (endorsed by HPS). Table 8 for Peach Bottom and Table 9 show the results of sensitivity calculations for dose truncation values for background and the Health Physics Society position compared with LNT results. Using these truncation values makes the already small latent cancer fatality risk estimates even smaller, in some cases by orders of magnitude. For Peach Bottom scenarios, the background results in Table 8 are the same as the HPS results because both truncation values exceed the plant-specific population return criterion of 0.5 rem/year. For Surry scenarios except ISLOCA, the background results in ~~Table 8~~ 9? differ from the HPS results because both truncation values do not exceed the plant-specific population return criterion of 4 rem over 5 years which is a simplification of Virginia's criteria of 2 rem in the first year and 0.5 rem/year in subsequent years. The ISLOCA results are the same to one significant digit within a radius of 10 miles for both truncation values because most of the emergency-phase doses exceed both of these criteria, while on the other hand, long-term doses are below these truncation levels or at least make an insignificant contribution to the overall doses. Using the 10 mrem/year truncation value made a relatively small change in the latent cancer risk compared with the LNT model and, therefore, these results were not included in Table 8 and Table 9. The results in Table 8 and Table 9 assume the release is truncated at 48 hours (72 hours for Surry long-term SBO) and the probability of 10 CFR 50.54(hh) mitigation is zero.

SOARCA analysis included predictions of individual latent cancer fatality risk for 2 distance intervals, 0 to 10 miles and 0 to 50 miles. The analysis indicated that individual latent cancer risk estimates generally decrease with increasing distance in large part due to plume dispersion and fission product deposition closer to the site.

As noted above, the SOARCA ~~offsite consequence~~ early fatality estimates are smaller than reported in earlier studies. For example, the Siting Study predicted 92 early fatalities for Peach Bottom and 45 early fatalities for Surry for the SST1 source term. In contrast, SOARCA predicted that the early fatality risk was essentially zero for both sites. For latent cancer fatality results, the exact basis for the Siting Study estimates could not be recovered, but literature searches and sensitivity analyses with MACCS2 suggested that these estimates are for the population within 500 miles of the site. ~~However~~ Therefore, given this uncertainty, SOARCA does not make a direct comparison to the Siting Study and NUREG-1150 latent cancer fatality estimates.

Also, because of the method used in NUREG-1150 to determine source terms, comparison of LCFs for that study are uncertain.

Table 8 Peach Bottom absolute Results for Scenarios without Successful Mitigation for LNT and Alternative Dose Response Models

Scenario	Absolute risk of latent cancer fatality for an individual located within 10 miles (per reactor-year)		
	Linear No Threshold	US **Background**	**Health Physics Society**
Long-term SBO	$6x10^{-10}$	$5x10^{-12}$	$5x10^{-12}$
Short-term SBO	$7x10^{-11}$	$3x10^{-12}$	$3x10^{-12}$

we need to put in all the results 0-50mi + the seismic sensitivity results -

Table 9 Surry absolute Results for Scenarios Without Successful Mitigation for LNT and Alternative Dose Response Models

Scenario	Absolute risk of latent cancer fatality for an individual located within 10 miles (per reactor-year)		
	Linear No Threshold	US **Background**	**Health Physics Society**
Long-term SBO	$7x10^{-10}$	$6x10^{-12}$	$2x10^{-14}$
Short-term SBO	$1x10^{-10}$	$5x10^{-12}$	$2x10^{-14}$
Thermally induced steam generator tube rupture	$1x10^{-10}$	$3x10^{-11}$	$5x10^{-12}$
Interfacing systems LOCA	$2x10^{-11}$	$8x10^{-12}$	$8x10^{-12}$

6.3 Conclusions

if we absolutely have to leave in something of the first sentence, at least take this out.

~~The results of the SOARCA project represent a major change from the way people perceive severe reactor accidents and their likelihood and consequences~~. Specific conclusions of the project are as follows:
for the sites and specific scenarios studied

- Mitigation is likely for all of the selected important scenarios due to time available for operator actions, and redundancy and diversity of equipment. Mitigation also resulted in no core damage for all scenarios except for the Surry STSBO and the Surry thermally induced steam generator tube rupture (TI-SGTR).
    - For the Surry STSBO with mitigation there was no containment failure within 48 hrs.
    - For the Surry TI-SGTR, the predicted individual latent cancer risk for the EPZ was small, $1x10^{-10}$ per reactor-yr, assuming LNT.

- If the scenarios were assumed to proceed unmitigated, i.e., leading to core damage, the best estimate MELCOR analyses indicated that accidents would progress more slowly and with smaller releases than past treatments (e.g., 1982 Siting Study) generally indicated. Large, early releases were not predicted because several of the phenomena (alpha failure, large scale direct containment heating and energetic hydrogen deflagration phenomena) that resulted in early containment failure in past studies are believed to be significantly less likely than in those earlier studies
- ~~The~~ unmitigated versions of the scenarios analyzed ~~have lower frequency, lower consequences, and lower risk than estimated in previous studies.~~

for

  - Individual early fatality risk is essentially zero
  - Individual latent cancer risk from the selected specific, important scenarios is thousands of times lower than the NRC Safety Goal and millions of times lower than all other cancer risks, even assuming the LNT dose response model
  - Using a dose response model which truncates annual doses below normal background levels results in a further reduction to the latent cancer risk, (by a factor of 100 for smaller releases and a factor of 3 for larger releases)

- Latent cancer fatality predictions are generally dominated by long term exposure to small annual doses (~500 mrem), in conjunction with return criteria for calculations using the LNT assumption.
- Bypass events do not pose higher latent cancer risk, higher conditional risk is offset by lower frequency.
- Explicit consideration of seismic impacts on emergency response (e.g., loss of bridges, traffic signals and delayed notification) did not significantly impact risk predictions
- The dominance of external events suggests the need for a corresponding PRA focus and seismic research.

~~Results, while specific to the two nuclear plants selected, may be generally applicable for plants with similar designs. However, results may vary depending on individual plant designs and capabilities and effectiveness of emergency response for surrounding populations.~~

7.0 REFERENCES

[1] WASH-740, *WASH-740: Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants*. 1957, Atomic Energy Commission: Washington, DC.

[2] WASH-1250, *WASH-1250: The Safety of Nuclear Power Reactors (Light Water-Cooled) and Related Facilities*. 1973, Atomic Energy Commission: Washington, DC.

[3] WASH-1400, *WASH-1400: Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants*. 1975, U.S. Nuclear Regulatory Commission: Washington, DC.

[4] NUREG/CR-2239, *Technical Guidance for Siting Criteria Development*. 1982, Sandia National Laboratories: Albuquerque, NM.

[5] NUREG-1150, V., *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants*. 1990, Nuclear Regulatory Commission: Washington, DC.

[6] *Safety Goals for the Operation of Nuclear Power Plants*, in *Federal Register, 51 FR 28044*. 1986.

[7] RG-1.174, *An approach for using probabilistic risk assessment in risk-informed decisions on plant-specific changes to the licensing basis*. 2002, Nuclear Regulatory Commission: Washington, DC.

[8] *Requirements for monitoring the effectiveness of maintenance at nuclear power plants.*, in *U.S. Code of Federal Regulations, 10 CFR Part 50.65*. 1999.

[9] *Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants.*, in *U.S. Code of Federal Regulations, 10 CFR Part 50.62*. 1984.

[10] *Loss of all alternating current power.*, in *U.S. Code of Federal Regulations, 10 CFR Part 50.63*. 1988.

[11] NUREG/CR-4334, *NUREG/CR-4334: An Approach to the Quantification of Seismic Margins in Nuclear Power Plants*. 1985, Lawrence Livermore National Laboratory: Livermore, CA.

[12] DiNunno, J.J. and e. al., *TID-14844: Calculation of Distance Factors for Power and Test Reactor Sites*. 1962, U.S. Atomic Energy Commission: Washtington, DC.

[13] Soffer, L., et al., *NUREG-1465: Accident Source Terms for Light-Water Nuclear Power Plants*. 1995, U.S. Nuclear Regulatory Commission: Washington, DC.

[14] Silberberg, M., et al., *NUREG-0956: Reassessment of the Technical Bases for Estimating Source Terms*. 1986, U.S. Nuclear Regulatory Commission: Washington, DC.

[15] *Determination of exclusion area, low population zone, and population center distance*, in *U.S. Code of Federal Regulations, 10 CFR Part 100.11*. 1962.

[16] Wall, I.B., et al., *NUREG-0340: Overview of the Reactor Safety Study Consequence Model.* 1977, U. S. Nuclear Regulatory Commission: Washington, DC.

[17] NUREG-0772, *NUREG-0772: Technical Bases for Estimating Fission Product Behavior During LWR Accidents.* 1981, U.S. Nuclear Regulatory Commission: Washington, DC.

[18] NUREG-0771, *NUREG-0771, Draft for Comment: Regulatory Impact of Nuclear Reactor Accident Source Term Assumption.* 1981, U.S. Nuclear Regulatory Commission: Washington, DC.

[19] Blond, R., et al., *NUREG-0773: The Development of Severe Accident Source Terms: 1957-1981.* 1982, U.S. Nuclear Regulatory Commission: Washington, DC.

[20] Gieske, J.A., et al., *BMI-2104: Radionuclide Release Under Specific LWR Accident Conditions.* 1985, Battelle Memorial Insitute: Columbus, OH.

[21] Boyack, B.E., et al., *LA-12240: MELCOR Peer Review.* 1992, Los Alamos National Laboratory: Los Alamos, NM.

[22] *Standards for combustible gas control system in light-water-cooled power reactors*, in *U.S. Code of Federal Regulations, 10 CFR Part 50.54.* 2003.

[23] Magallon, D., I. Huhtiniemi, and H. Hohmann, *Lessons learned from FARO/TERMOS corium melt quenching experiments.* Nuclear Engineering and Design, 1999. **189**: p. 223-238.

[24] Chu, T.Y., et al., *NUREG/CR-5582, SAND98-2047: Lower Head Failure Experiments and Analyses.* 1998, Sandia National Laboratories: Albuquerque, NM.

[25] Farmer, M.T., S. Lomperski, and S. Basu. *Results of Reactor Material Experiments Investigating 2-D Core-Concrete Interaction and Debris Coolability.* in *International Conference on Advanced Power Plants, ICAPP'04.* 2004. Pittsburgh, Pennsylvania.

[26] Carbajo, J.J., *NUREG/CR-5942, ORNL/TM-12229: Severe Accident Source Term Characteristics for Selected Peach Bottom Sequences Predicted by the MELCOR code.* 1993, Oak Ridge National Laboratory: Oak Ridge, TN.

[27] Bayless, P.B., *NUREG/CR-5214, EGG-2547: Analysis of Natural Circulation During A Surry Station Blackout Using SCDAP/RELAP5.* 1988, Idaho National Engineering Laboratory: Idaho Falls, ID.

[28] NEA/CSNI, *NEA/CSNI/R(99)24: Technical Opinion Paper on Fuel-Coolant Interaction.* 2000, Nuclear Energy Agency Committee on the Safety of Nuclear Installations, Organization for Economic Cooperation and Development: Paris, France.

[29] Pilch, M.M. and T.G. Theofanous, *The probability of containment failure by direct containment heating in Zion.* Nuclear Engineering and Design, 1996. **164**: p. 1-36.

[30] Theofanous, T.G., et al., *NUREG/CR-6025: The Probability of Mark-I Containment Failure by Melt-Attack of the Liner.* 1993, U. S. Nuclear Regulatory Commission: Washington, DC.

[31] Gauntt, R.O., et al., *NUREG/CR-6119, Vol., Rev. 3, MELCOR Computer Code Manuals, Vol. 1: Primer and User's Guide, Version 1.8.6*. 2005, Sandia National Laboratories: Albuquerque, NM.

[32] Lorenz, R.A. and M.F. Osborne, *NUREG/CR-6261: A Summary of ORNL Fission Product Release Tests with Recommended Release Rates and Diffusion Coefficients*. 1995, U. S. Nuclear Regulatory Commission: Washington, DC.

[33] Clement, B. and T. Haste, *Comparison report on International Standard Problem ISP-46 (Phebus FPT-1)*. 2003, Note Technique SEMAR 03/021, Draft Final Report.

[34] Ducros, G., et al., *Fission product release under severe accidental conditions: general presentation of the program and synthesis of VERCORS 1-6 results*. Nuclear Engineering and Design, 2001. **208**: p. 191-203.

[35] Greene, N.M., L.M. Petrie, and R.M. Westfall, *ORNL/TM-2005/39, Version 5, Vols. I-III: NITAWL-III: Scale System Module for Performing Resonance Shielding and Working Library Production, SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations*. 2005, Oak Ridge National Laboratory: Oak Ridge, TN.

[36] ORNL/TM-2005/39, *ORNL/TM-2005/39, Version 5.1., Vols. I-III: SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations*. 2006, Oak Ridge National Laboratory: Oak Ridge, TN.

[37] RG-3.54, R., *Spent Fuel Heat Generation in an Independent Spent Fuel Installation*. 1999, U.S. Nuclear Regulatory Commission: Washington, DC.

[38] DeHart, M.D. and S.M. Bowman. *Improved radiochemical assay analyses using TRITON depletion sequences in SCALE*. in *Proceedings of International Atomic Energy Agency Technical Meeting "Advances in Applications of Burnup Credit to Enhance Spent Fuel Transportation, Storage, Reprocessing and Disposition*. 2005. London, United Kingdom: International Atomic Energy Agency.

[39] Bowman, S.M. and D.F. Gill. *Validation of Standardized Computer Analyses for Licensing Evaluation/TRITON Two-Dimensional and Three-Dimensional Models for Light Water Reactor Fuel*. in *PHYSOR-2006, American Nuclear Society Topical Meeting on Reactor Physics: Advances in Nuclear Analysis and Simulation*. 2006. Vancouver, British Columbia, Canada: American Nuclear Society.

[40] Germina, I. and I.C. Gauld, *Analysis of Decay Heat Measurements for BWR Fuel Assemblies*. Transactions of the American Nuclear Society, 2006. **94**: p. 385-387.

[41] Chanin, D. and M.L. Young, *NUREG/CR-6613, SAND97-0594: Code Manual for MACCS2 User's Guide*. 1997, Sandia National Laboratories: Albuquerque, NM.

[42] NUREG-0917, *NUREG-0917: Nuclear Regulatory Commission Staff Computer Programs for Use With Meteorological Data*. 1982, U.S. Nuclear Regulatory Commission: Washington, DC.

[43] RG-1.23, *Regulatory Guide 1.23, Rev. 1: Meteorological Monitoring Programs for Nuclear Power Plants*. 2007, U. S. Nuclear Regulatory Commission: Washington, DC.

[44] NUREG-0654, *NUREG-0654/FEMA-REP-1, Rev. 1: Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants*. 1979, U.S. Nuclear Regulatory Commission: Washington, DC.

[45] NUREG/CR-6864, *NUREG/CR-6864: Identification and Analysis of Factors Affecting Emergency Evacuations*. 2005, Sandia National Laboratories: Albuquerque, NM.

[46] Oak Ridge National Laboratory, *Oak Ridge Evacuation Modeling System (OREMS)*. 2003, Oak Ridge National Laboratory: Oak Ridge, TN.

[47] McFadden, K.L., N.E. Bixler, and R.O. Gauntt, *MELMACCS System Documentation (MELCOR to MACCS2 interface definition)*. 2005, Sandia National Laboratories: Albuquerque, NM.

[48] Bixler, N.E., E. Clause, C. W. Morrow, and J. A. Mitchell, *Evaluation of Distributions Representing Important Non-Site Specific Parameters in Off-Site Consequence Analyses*, to be published as a NUREG Report, Sandia National Laboratories: Albuquerque, NM, 2010.

[49] Bixler, N.E., et al., *NUREG/ER-6525, Rev. 1, SAND2003-1648P: SECPOP2000: Sector Population, Land Fraction, and Economic Estimation Program*. 2003, Sandia National Laboratories: Albuquerque, NM.

[50] EPA (2202) Federal Guidance Report No. 13 CD Supplement, E.-C.-.-., Rev. 1. 2002, Prepared by Oak Ridge National Laboratory, Oak Ridge, TN for Office of Air and Radiation, U. S. Environmental Protection Agency: Washington, DC.

[51] Eckerman, K. F., *Risk Coefficients for SOARCA Project*, memo, Oak Ridge National Laboratory, May 2008.

[52] NAS, *Health Effects of Exposure to Low Levels of Ionizing Radiaiton: BEIR V*. 1990, National Academy of Sciences, National Research Council, National Academy Press: Washington, DC.

[53] Aurengo, A., et al., *French National Academy of Medicine report: Dose-effect relationships and estimation of the carcinogenic effects of low doses of ionizing radiation*. 2005, French Academy of Sciences, French National Academy of Medicine.

[54] ICRP, *The 2007 Recommendations of the International Commission on Radiological Protection*. Annals of the ICRP, 2007. **37**(Nos. 2-4).

[55] ICRP, *Scope of Radiological Protection Control Measures*. Annals of the ICRP, 2007. **37**(Nos. 5).

[56] HPS, *PS010-1: Position Statement of the Health Physics Society -- Radiation Risk in Perspective*. 2004, Health Physics Society: McLean, VA.

[57] NUREG-6953, Volume 2, "Review of NUREG-0654, Supplement 3, 'Criteria for Protective Action Recommendations for Severe Accidents' - Focus Group and Telephone Survey," Nuclear Regulatory Commission, Washington, DC, 2008

[58] NUREG/CR-6244, Volume 1, "Probabilistic Accident Consequence Uncertainty Analysis, Dispersion and Deposition Uncertainty Assessment, Main Report," U.S. Nuclear Regulatory Commission, Washington, DC, 1995

[59] NUREG/CR-6545, Volume 1, "Probabilistic Accident Consequence Uncertainty Analysis, Early Health Effects Uncertainty Analysis, Main Report," U.S. Nuclear Regulatory Commission, Washington, DC, 1997

[60] Laur, M.N., "Meeting with Sandia National Laboratories and an Expert Panel on MELCOR/MACCS Codes in Support of the State of the Art Reactor Consequence Analysis Project," Memo to J.T. Yerokun, Agency Document Access and Management System Accession Number ML062500078, U.S. Nuclear Regulatory Agency, Washington, DC, September, 2006.

[61] NUREG/CR-7008, "Best Practices for Simulation of Severe Accident Progression at Nuclear Power Plants," Nuclear Regulatory Commission, Washington, DC, 2010 (To be published)