UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

June 16, 2010

MEMORANDUM TO:	William J. Shack, Chairman Regulatory Policies and Practices Subcommittee
FROM:	Hossein Nourbakhsh, Senior Technical Advisor */RA/*
SUBJECT:	STATUS REPORT FOR THE MEETING OF THE SUBCOMMITTEE ON REGULATORY POLICIES AND PRACTICES, JUNE 21, 2010, IN ROCKVILLE, MARYLAND

The purpose of this memorandum is to forward written materials for your use in preparing for the meeting of the ACRS Subcommittee on Regulatory Policies and Practices on June 21, 2007. The Subcommittee will discuss the status of staff's efforts associated with the State-Of-the-Art Reactor Consequence Analysis (SOARCA) Project. The purpose of the meeting is to gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee. Attached please find the agenda, status report, and background materials.

Attendance by the following members is anticipated and reservations have been made at the following hotels for June 20-21, 2010, unless otherwise indicated.

Shack	Residence Inn	Bley	None (Local)
Corradini	Residence Inn	Ryan	Bethesda N. Marriott
Stetkar	Bethesda N. Marriott		

Please notify ACRS Travel@nrc.gov if you need to change or cancel the above reservations.

Attachments[1]
1. Agenda
2. Status report
3. Supporting Documents
   I. The SOARCA Summary draft NUREG Rev. 2a
   II. Appendix A to the SOARCA draft NUREG Rev. 2
   III. Appendix B to the SOARCA draft NUREG Rev. 2
   IV. The Peer Review individual reviewer draft reports

cc:	ACRS Members	
cc w/o attach:	E. Hackett	
	C. Santos	M. Afshar-Tous

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1 Electronic copies of the supporting documents have been sent to members separately.

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Attachment 1

Advisory Committee on Reactor Safeguards
Regulatory Policies and Practices Subcommittee Meeting
Rockville, MD
June 21, 2010

- Proposed Agenda –
(CLOSED)
Cognizant ACRS Staff: Hossein Nourbakhsh (301-415-5622, hpn@nrc.gov)

	Topic	Presenter(s)	Time
	Opening Remarks and Objectives	W. Shack, ACRS	8:30-8:40 am
I	Background, Objectives, and Conclusions	C. Tinkler, RES	8:40-9:45 am
II	Scenario Selection	M. Stutzke, RES	9:45-10:15 am
	Break		10:15-10:30am
III	Mitigating Measures Assessment	R. Prato, NRO	10:30am-11:15am
	Emergency Preparedness	R. Sullivan, NSIR	11:15am-12:00 pm
	Lunch		12:00-12:45pm
IV	Accident progression and Source Term Analysis	J. Schaperow, RES	12:45-2:00 pm
V	Offsite Consequence Analysis	J. Mitchell, RES	2:00- 2:45 pm
	Break		2:45-3:00 pm
VI	Closing Remarks	J. Yerokun, RES	3:00-3:15 pm
VII	Discussion	ALL	3:15-4:15pm
	Adjourn		4:15 pm

Notes:
• During the meeting, use 301-415-7360 to contact anyone in the ACRS Office.

• Presentation time should not exceed 50 percent of the total time allocated for a given item. The remaining 50 percent of the time is reserved for discussion.

•Thirty five (35) hard copies of each presentation or handout should be provided to the Designated Federal Official (DFO) / ACRS Contact 30 minutes before the meeting.

•One (1) electronic copy of each presentation should be e-mailed to the DFO / ACRS Contact 1 day before the meeting. If an electronic copy cannot be provided within this timeframe, presenters should provide the DFO / ACRS Contact with a CD containing each presentation at least 30 minutes before the meeting.

Attachment 2

~~FOR INTERNAL ACRS USE ONLY~~

Advisory Committee on Reactor Safeguards
Subcommittee on Regulatory Policy and Practices
Rockville, MD
June 21, 2010

- Status Report -

PURPOSE

The Subcommittee will discuss the status of the staff's efforts associated with the State-Of-the-Art Reactor Consequence Analysis (SOARCA) project. The purpose of the meeting is to gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

BACKGROUND

The phenomenology and offsite consequences of severe reactor accidents have been the subject of considerable research by the NRC. Over the years, several systematic attempts have been made to use quantitative techniques to estimate the probabilities, source terms, and public consequences from potential accidents in commercial nuclear power plants. The Reactor Safety Study (WASH-1400) was the first systematic attempt to provide estimates of public risk. This 1975 study included analytical methods for determining both the probabilities and consequences of various accident scenarios. Two specific reactor designs were analyzed in WASH-1400: Peach Bottom Atomic Power Station, a Boiling Water Reactor (BWR) with a Mark I containment and Surry, a 3-loop Westinghouse Pressurized Water Reactor (PWR) with a subatmospheric containment.

Sandia National Laboratory (SNL) performed a study of technical aspects of siting for nuclear power reactors. The results of this study, also known as Sandia Siting Study, were published in NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," December 1982. This study used five generic source terms for analyzing the consequences and socio-economic impacts of possible plant accidents at 91 existing or proposed reactor sites. These source terms were derived from the Reactor Safety Study (WASH-1400) and its immediate successors.

Since the publication of the Sandia Siting Study, many events have brought a new focus to this study and its results. The results, in terms of predicted offsite early fatalities and latent cancer, have often been quoted by outside organizations to illustrate the potential consequences of a severe accident at a commercial nuclear power plant. Despite accepted arguments that these results does not present an up-to-date picture of consequences at nuclear power plants and does not reflect current state-of-the-art in evaluating severe accident progression and offsite consequences.

On request from the Commission, the staff sent forward to the Commission a paper describing a proposed plan for developing state-of-the-art reactor consequence analyses for all commercial nuclear power plant sites. The Commission responded in an April 14, 2006 Staff Requirements Memorandum (SRM) with a general approval of the plan. The Commission directed the staff to "use the improved understanding of source terms and severe accident phenomenology (e.g., containment failure modes, time of release, release duration, inventory release fractions), and credit the use of Severe Accident Management Guidelines (SAMGs) and other new procedures, such as mitigative measures resulting from B.5.b and other like programs, that were not in place when the earlier study was performed." The Commission also instructed the staff to "present its updated results using risk communication techniques to achieve an informed public understanding of the extent and value of defense-in-depth features including current mitigative strategies, and of the important analytical assumptions."

In the April 14, 2006 SRM, the Commission specifically instructed the staff to "work with the ACRS on technical issues such as identification of accident scenarios to be evaluated, evaluation of source terms, credit for operator actions or plant mitigation systems, modeling of emergency preparedness, modeling of offsite consequences, and definition and characterization of analysis uncertainty."

In an April 2, 2007 SRM, the Commission directed the staff to "reduce the initial scope of this effort to not more than eight plants representing a spectrum of plant vendors and technologies." The Commission also directed the staff to "conduct the first assessments on a subset of the eight plants, for example a selected BWR and PWR plant, in order to resolve issues associated with the integration of methods and resolve details associated with simulation of plant systems and procedures." The Commission also instructed the staff to "provide the results of these studies to the Commission along with a recommendation, based on the insights gained from the initial eight studies, as to whether continuing this project as originally described in SECY-05-0233 is necessary to achieve its objectives."

During the 535th meeting of the ACRS, September 7-9, 2006, the staff briefed the Committee on its plan for the state-of-the-art consequence analyses project. During the 538th meeting of the ACRS, December 6-8, 2006, the Committee discussed the status of staff's effort associated with the SOARCA project. The staff briefed the committee on a number of topics related to this project including plans for MELCOR and MACCS code improvement, selection of scenarios to use for consequence analysis. The staff also briefed the Committee on its plan for a site-specific simulation of offsite emergency response for this project. The Members had many questions regarding the technical details of this study and how uncertainties will be addressed. The Members agreed that the technical details be discussed in a subcommittee as the process and calculations further develops.

The ACRS Subcommittee on Regulatory Policies and Practices held a meeting on July 10, 2007 to discuss the status of staff's efforts associated with the State-of-the-Art Reactor Consequence Analysis (SOARCA) Project. During the meeting, the Subcommittee reviewed several topics including accident sequence selection, containment system states, MELCOR analysis, emergency preparedness, and MACCS2 analysis. As directed by the Commission, the staff has reduced the initial scope of SOARCA Project. The staff is initially focusing on two sites, Peach Bottom in Pennsylvania, and Surry in Virginia. During the closed portion of the Subcommittee meeting, the Subcommittee discussed the staff's initial findings of the accident sequence selection, preliminary MELCOR insights, containment performance, and emergency preparedness for these two plants. The Subcommittee also discussed the various options that staff is evaluating for assessment of dose thresholds for latent cancer fatalities. The members agreed to continue their review of SOARCA project in a subcommittee meeting as the staff makes further progress in its analysis.

The ACRS Subcommittee on Regulatory Policies and Practices also held a meeting on November 16, 2007 to discuss the current status of staff's efforts associated with the SOARCA project. During the meeting, the staff presented some preliminary results of SOARCA for Peach Bottom and Surry Plants. During the 548th meeting of the ACRS, December 6-8, 2007, the Committee discussed the status of staff's effort associated with the SOARCA project. During the 549th meeting of ACRS, February 7-9, 2008, the Committee completed its initial review of the staff's activities regarding the SOARCA Project. In its February 25, 2008 report to the Commission concerning the SOARCA Project [12], the ACRS recommended that "as a minimum, a limited set of updated Level-3 PRAs for the SOARCA pilot plants be performed to benchmark the consequence analyses and provide useful information to the Commission in deciding whether to proceed with a full set of consequence analyses." The ACRS further noted that "examination of the Level-3 PRA results for the SOARCA pilot plants may identify suitable Level-1 event scenario screening criteria and simplifying assumptions that could be used to develop a defensible, simplified approach." The Committee also recommended that "the process for selecting the external event sequences in SOARCA needs to be made more comprehensive. The impacts from these events on containment mitigation systems, operator actions, and offsite emergency responses should be evaluated realistically." The Committee further recommended that "consequences be expressed in terms of ranges calculated using the threshold recommended by the Health Physics Society Position Statement and some lower thresholds. A calculation with linear, no-threshold (LNT) should also be performed, which would facilitate comparison with historical results."

In a letter dated April 7, 2008, the Executive Director for Operations (EDO) responded to the ACRS report of February 25, 2008 on SOARCA, indicating that the staff did not agree with the ACRS recommendation that a limited set of Level-3 PRAs be performed to benchmark the SOARCA approach developed by the staff. In its April 21, 2008 response to the EDO, the ACRS noted that the Committee continues "to believe that the credibility of the SOARCA Project cannot rely on confidence in the judgment of the staff and on a novel analysis procedure that differs substantially from previous state-of-the-art analyses of the consequences of severe reactor accidents." The ACRS further noted that "without including benchmark analyses similar in scope, it will be difficult to demonstrate convincingly that reductions in consequences that might be indicated by the SOARCA results reflect the impact of enhancements in plant design and operation, and improvements in calculation methods for accident progression and consequence analysis, rather than changes in the scope of the calculation."

In a June 26, 2008 SRM, resulting from the June 5, 2008 meeting with the ACRS, the Commission directed the staff to "continue working to address Committee concerns, such as with SOARCA."

To support its continued dialogue with the staff on SOARCA, The Committee asked ACRS Senior Technical Advisor, to prepare white paper on "Historical Perspectives and Insights on Reactor Consequence Analyses". This paper, which was transmitted to EDO as an attachment to a November 14, 2008, ACRS Letter, provides an overview of previous major studies of reactor consequences and concludes it is feasible to develop a simplified approach for updating results from earlier Level-3 PRAs such as NUREG-1150 for comparison with aspects of SOARCA results.

In a September 10, 2008 SRM regarding SECY-08-0029, "State-Of-the-Art Reactor Consequence Analysis -- Reporting Offsite Health Consequences," the Commission approved the staff's Recommendation 6 for assessing and reporting offsite health consequences in the state-of-the-art reactor consequence analyses (SOARCA) project. The Staff's Recommendation 6 as noted in SECY-08-0029 is:

> Calculate the average individual likelihood of an early fatality and LCF that is expressed as the average probability of a population-weighted, average individual (age and gender averaged) dying from cancer conditional to the occurrence of a severe reactor accident. The calculation would include both LNT and 100 μSv (10 mrem) dose response models, with results presented for three distances: (1) 0 to 16.1 km (10 miles); (2) 0 to 80.5 km (50 miles); and (3) 0 to 161 km (100 miles).

In September 10, 2008 SRM, the Commission also approved the staff's recommendation to submit the Peach Bottom and Surry methodology and approaches for peer review by a cadre of experts who have not participated in the development of the SOARCA and who have expertise in one or more areas of the disciplines employed in the SOARCA. The Commission directed the staff that "the peer review should address, in part, the credibility of the underlying assumptions and engineering judgment employed by the staff in the performance of these consequence analyses and whether it is necessary to perform level-3 PRAs to corroborate the integrity of the SOARCA Project."

A first draft of the SOARCA NUREG document was completed in July 2009. The SOARCA Peer Review Committee was appointed to provide an independent review of the analyses. Technical experts (11 members) from industry, consulting, academia, and research laboratories have been assembled to assess all aspects of the project and provide guidance and suggestions. The Peer Review Committee began its work in July 2009 and submitted its draft peer review report in May 2010.

The purpose of this Subcommittee meeting is to discuss the draft NUREG-1935, "State-Of-the-Art Reactor Consequence Analyses (SOARCA) Project," as well as the draft peer review report, "Peer Review of the State-Of-the-Art Reactor Consequence Analysis (SOARCA) Project."

DISCUSSIONS

The staff is initially focusing on two sites, Peach Bottom in Pennsylvania, and Surry in Virginia. These two plants have been the subject of a number of earlier risk studies including the

NUREG-1150, “Severe Accident Risk - An Assessment of Five U. S. Nuclear Power Plants.” The results from these studies have provided valuable insights for the initial phase of SOARCA project.

There is an inconsistency in characterizing SOARCA throughout both NUREG-1935 and peer review reports. SOARCA has been referred to, for example, as:

> “best estimate evaluation of the likely consequences of important severe accident events at reactor sites in the U.S. civilian nuclear power reactor fleet,” [Page iii, NUREG-1935]
>
> “updated estimates of potential site-specific offsite consequences from severe accidents for operating nuclear power plants (NPPs),” [Page 1, 3rd Paragraph, NUREG-1935]
>
> “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,” [Page 1, 3rd Paragraph, NUREG-1935]
>
> "an updated reference of the likely outcomes of severe reactor accidents," [Page 1, 4th Paragraph, NUREG-1935]
>
> analysis of “a set of important accident sequences considering both likelihood and potential consequences,” [Page 7, 3rd Paragraph, NUREG-1935]
>
> “risk-informed consequence analyses,” [Page 1, 3rd Paragraph, NUREG-1935]
>
> “update[d] evaluations of hypothetical severe accident progression and offsite consequences in nuclear reactors,” [Page 1, 1st Paragraph, Peer Review Report]
>
> “more realistic assessments of the risks posed by nuclear power plants” [Page 1, 1st Paragraph, Peer Review Report], and
>
> “assessment of health consequence risks from severe accidents [Page 26, 1st paragraph, NUREG-1935.]

An overview of the SOARCA process is depicted in Figure 1. Major elements of SOARCA include sequence selection, determining containment systems availability status, assessment of mitigative measures, structural analysis and evaluation of containment performance, severe accident progression analyses using MELCOR computer code to determine source term, modeling the protective response afforded by the current site-specific Emergency Preparedness (EP) programs, and performing consequence analyses using MACCS2 computer code to determine early fatalities (EF), and latent cancer fatalities (LCF).

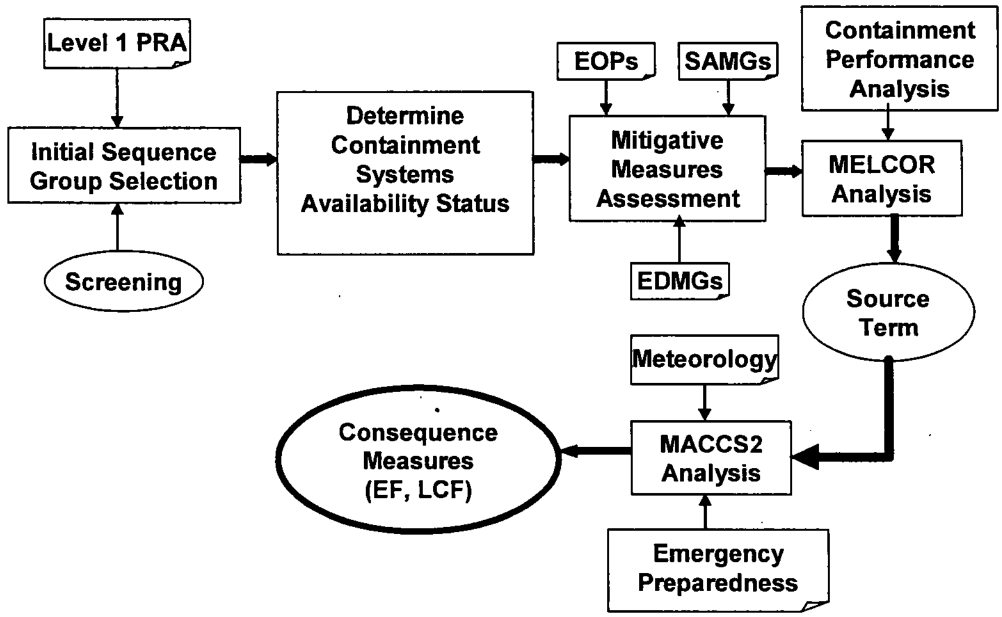

Figure 1 SOARCA Process

Accident Sequence Group Selection

Plant specific SPAR models and licensee PRA results are used to identify the internal event core damage sequences. Once the sequences determined, they are grouped based on the similarity in the availability of front-line systems and on the timing of the sequences. An initial screening is then performed to identify sequence groups with a CDF > 10^{-6} per reactor year for most sequence groups and CDF>10^{-7} per reactor year for sequence groups that are known to have the potential for higher consequences (e.g. containment bypass sequences such as steam generator tube rupture and interfacing system LOCA initiators). For external events, available plant/design-specific assessments (NUREG-1150, IPEEE, etc.), external event SPAR (SPAR-EE) model, and/or generic insights are used to define and select representative groups.

The internal events assessment, using SOARCA screening criteria, resulted in retention of no internal event sequences for Peach Bottom, and only 2 containment bypass sequences (Spontaneous Steam Generator Tube Rupture, and Interfacing Systems LOCA in the LPI System) for Surry. However, for both plants, representative external event sequences were identified. For Peach Bottom, Two accident scenario (both initiated by a seismic event) were selected: (1) Seismic-Initiated Long-Term Station Blackout, and (2) Seismic-Initiated Short-Term Station Blackout. For Surry, two external event scenarios were selected: (1) Seismic-Initiated Long-Term Station Blackout, and (2) Seismic-Initiated Short-Term Station Blackout. It should be noted that extreme seismic events which involve failure of the containment and lead to core damage have been excluded. The staff has concluded 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.

The approach and basis for the selection of chosen sequences, particularly from external events, is not defensible. A more objective approach to sequence group selection including consideration of uncertainties is needed. For example, draft NUREG-1935 (page 15, 3rd paragraph) states that "the selected [external events] 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." However, no basis for such conclusion and judgment has been provided. For flood and fire scenarios, the consequences of the flood or fire may prevent certain mitigative measures from taking place but they are less likely to be affected by the seismic events considered in SOARCA.

Determining Containment Systems Availability Status

Simple approaches was used to determine the availability of systems not considered in level 1, but are important for the containment accident progression and radionuclide release. 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 SPAR system models (fault trees). In this approach random failures and human errors associated with the containment systems were neglected. It is not clear how judgments were made on system availability for external event initiated sequences.

Mitigative Measures Assessment

For each sequence groupings within the scope of the site specific analyses, applicable mitigative measures that can potentially prevent or delay core damage, RCS failure, and/or containment failure are identified. In addition the approximate time for implementation of such measures after the initiating event are estimated for input into the MELCOR analysis. 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.

The assessment of mitigative measures is based on the licensees' 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.

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. The staff conclusion that there is some conservatism in attributing all of the external event likelihood to a seismic initiator may be questionable.

Accident Progression and Source Term Analyses

Severe accident MELCOR code is used for accident progression analyses for each plant to determine radionuclide release characteristics (source term) into environment as input in the consequence analyses.

The staff has performed structural analyses for predicting the performance of steel (Mark I) and reinforced concrete (PWR) containment structures, in terms of leak rate or area versus pressure. These analyses are based on earlier studies including containment integrity research performed at Sandia national Laboratories (SNL), Individual Plant examination (IPE) reports, and NUREG-1150 study. These results are used for input for MELCOR analyses.

NUREG-1935 states that "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, source term, and offsite consequences. The panel also reviewed a set of code enhancements as well as consideration of the SOARCA project in general." NUREG-1935 refers to NUREG/CR-7008 (yet to be published) for discussions of the specific modeling practices. The state-of-knowledge uncertainties associated with the accident progression phenomena and source term analyses has yet to be addressed.

The results MELCOR analyses indicate that the time intervals between the start of core damage and bottom head failure are substantially longer than those predicted by earlier MELCOR analyses performed in 1990s. It is not clear how sensitive these results are to the number of nodes used in the MELCOR models. In view of significant impact of severe accident progression timing on mitigative measures assessment and potential health consequences, the uncertainties associated with severe accident progression timing should also be addressed.

Probabilistic risk assessments have shown that the largest contributors to overall plant risk are usually those accident sequences in which the containment is either bypassed or fails early. The probability of occurrence of this class of accidents tends to be lower than the probability of accidents in which the containment does not fail or is not bypassed. Since the offsite consequences of accidents leading to early containment failure (e.g., early containment rupture) and bypass are so much larger than accidents where the containment does not fail (e.g., $5X10^7$ person-rem vs. 250 person-rem), they usually dominate overall plant risk. Therefore in severe accident analyses in support of PRAs less attention has been given to the accident progression and source term issues that only impact the consequences of no containment failure accident sequences. However, because the high consequence accident sequences are screened out by SOARCA (due to their low frequencies) uncertainties associated with those issues may become more important. Examples of such issues are containment leak rate, chemical form of iodine, and late re-volatilization and release of volatile nuclides that were deposited within the reactor coolant system earlier during core degradation. Therefore some of the assumptions used In the MELCOR calculations for SOARCA may be questionable.

MACCS2 Analysis

MACCS2 is used for consequence analyses for each plant to determine early fatalities (EF), and latent cancer fatalities. A site specific model for each plant analyzed has been developed based on meteorological data and emergency response parameter.

A range of dose truncation values was considered in SOARCA, ranging from linear non threshold (LNT) assumption on one hand to the Health Physics Society recommendation (5 rem/yr and 10 rem lifetime) on the other hand. Two intermediate dose-truncation levels were also considered. One is the 10 mrem/yr dose truncation value suggested in ICRP Report 104; the other is US-average background radiation of 620 mrem/yr. Results for these four dose-truncation levels are reported without bias for each of the accident scenarios considered in the SOARCA study.

Mean, population-weighted, individual risks are presented from 0 to 50 miles. The 0-10 mile range represents the population within the EPZ. The term "population-weighted" carries the meaning of the effect of population distribution, along with wind rose probabilities, on the predicted risk. This statistic is simply the number of predicted fatalities divided by the population within a specified region. The use of the word "mean" is intended to convey that the results are arithmetic averages over the annual weather data used in the analysis. The initial phase of the SOARCA only considers uncertainty in the weather. The staff plans to perform uncertainty analyses considering the effect of source term and other input uncertainties on the predicted consequences in future.

In view of the major objective of SOARCA being to provide update quantification of offsite consequences found in earlier NRC publications such as NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," lack of any direct comparison of the results of SOARCA with the previous consequence analysis may be questionable.

Peer Review of SOARCA

The SOARCA Peer Review Committee were requested to assess the methodological approach, underlying assumptions, results and conclusions obtained for Peach Bottom and Surry reactors. A consensus opinion of the Committee was not pursued throughout the review process. All of the written comments which were provided to the SOARCA team by the reviewers, have been assembled by and coordinated through the Peer Review Committee chair.

The comments provided by the members of the Peer Review Committee are, in general, consistent with the views expressed previously by the ACRS. For example:

> *"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." (Ken Canavan)

"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." (Ken Canavan)

SOARCA being not a full level 3 PRA study, only a limited number of scenarios has been selected. (B. Clément)

"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." (B. Clément)

"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." (Jeff R. Gabor)

"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." (David Leaver)

"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." (David Leaver)

"It is the opinion of this reviewer that these objectives were only partially achieved. ..... 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." (Bruce Mrowca)

"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. " (Bruce Mrowca)
"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." (Bruce Mrowca)

SUBCOMMITTEE ACTION

The Subcommittee should be prepared to provide its views and recommendations to the Full Committee, at the October meeting. The Committee is expected to write a letter on SOARCA at that time.