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

November 10, 2007

MEMORANDUM TO: William J. Shack, Chairman
Subcommittee on Regulatory Policies and Practices

FROM: Hossein Nourbakhsh, Senior Staff Engineer **/RA/**

SUBJECT: STATUS REPORT FOR THE MEETING OF THE SUBCOMMITTEE ON REGULATORY POLICIES AND PRACTICES, NOVEMBER 16, 2007, 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 November 16, 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 November 15-17, 2007, unless otherwise indicated.

Shack	RESIDENCE INN	Corradini	RESIDENCE INN (15-16)
Abdel-Khalik	BETH. N. MARRIOTT	Stetkar	BETH. N. MARRIOTT (15-16)
Apostolakis	RESIDENCE INN (15-16)	Kress	Hilton (15-16)
Armijo	BETH. N. MARRIOTT (15-16)	Wallis	Hilton
Bley	NONE (LOCAL)		

Please notify Ms. Barbara Jo White at 301-415-7130 if you need to change or cancel the above reservations.

Attachments[1]
1. Agenda
2. Status report
3. Memorandum, Dated October 22, 2007, from Jimi T. Yerokun, Chief, Risk Applications and Special Projects Branch, Office of Nuclear Regulatory Research, to Cayotnana (Tanny) Santos, Chief, Nuclear Reactors Branch, ACRS, Subject: DOCUMENTS FOR ACRS SUBCOMMITTEE REVIEW OF SOARCA PROJECR

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

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Enclosures:

1. "Sequence Selection and Containment System Status Evaluation for Use in the State of the Art Reactor Consequence Analysis (SOARCA) Project"
2. Peach Bottom Internal Event Sequence Groups
3. Surry Internal Event Sequence Groups
4. Peach Bottom Dominant External Sequence Group Summary Containment System Status
5. Surry Dominant External Sequence Group Summary Containment System Status
6. Sequence Summaries
7. "Peer Review Of SOARCA Accident Sequence Screening"
8. "SOARCA Sequence Selection Peer Review Panel Comments and Responses to Comments"
9. Peach Bottom Sequence Selection Evaluation
10. An Assessment of Dominant External Event Sequences for Peach Bottom NPP
11. SOARCA Mitigative Measures Assessment for Peach Bottom
12. An Assessment of Containment Structural Behavior Due to Severe Internal Loading Conditions at Beach Bottom
13. Surry Sequence Selection Evaluation
14. An Assessment of Dominant External Event Sequences for Surry NPP
15. SOARCA Mitigative Measures Assessment for Surry
16. An Assessment of Containment Structural Behavior Due to Severe Internal Loading Conditions for Concrete Reinforced Containment Structures
17. SOARCA Project Formatting of Raw meteorological Data for MACCS2 Input

4. Memorandum, Dated November 1, 2007, from Jimi T. Yerokun, Chief, Risk Applications and Special Projects Branch, Office of Nuclear Regulatory Research, to Cayotnana (Tanny) Santos, Chief, Nuclear Reactors Branch, ACRS, Subject: ACRS MEETINGS ON STATE-OF-THE ART REACTOR CONSEQUENCE ANALYSES
Enclosure: "STATE-OF-THE ART REACTOR CONSEQUENCE ANALYSES," Commission Technical Assistants Briefing, Octobe 1, 2007, (Official Use Only)

cc: ACRS Members
cc w/o attach: F. Gillespie
C. Santos

Attachment 1

Advisory Committee on Reactor Safeguards
Regulatory Policies and Practices Subcommittee Meeting
Rockville, MD
November 16, 2007

- Proposed Agenda –

(CLOSED)

Cognizant Staff Engineer: 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	State-Of-the Art Reactor Consequence Analysis (SOARCA) Project Overview	R. Prato, RES	8:40-9:30 am
II	Structural Analysis	A. Istar, RES	9:30-10:30 am
	Break		10:30-10:45am
III	Peach Bottom Results	J. Schaperow, RES	10:45am-12:15pm
	Lunch		12:15-1:00 pm
IV	Surry Results	J. Schaperow, RES	1:00-2:45 pm
V	Dose Threshold	R. Prato	2:45- 3:15 pm
	Break		3:15-3:30 pm
VI	Path Forward	R. Prato	3:30-4:00 pm
VII	Discussion	ALL	4:00-5:00pm
	Adjourn		5:00 pm

Notes:

- Presentation time should not exceed 50% of the total time allocated for a specific item.
- Number of copies of presentation materials to be provided to the ACRS - 25.

Attachment 2

~~FOR INTERNAL ACRS USE ONLY~~

Advisory Committee on Reactor Safeguards
Subcommittee on Regulatory Policy and Practices
Rockville, MD
November 16, 2007

- 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 purpose of this Subcommittee meeting is to discuss the current status of the staff's efforts associated with this 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 should provide valuable insights for the initial phase of SOARCA project.

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).

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.

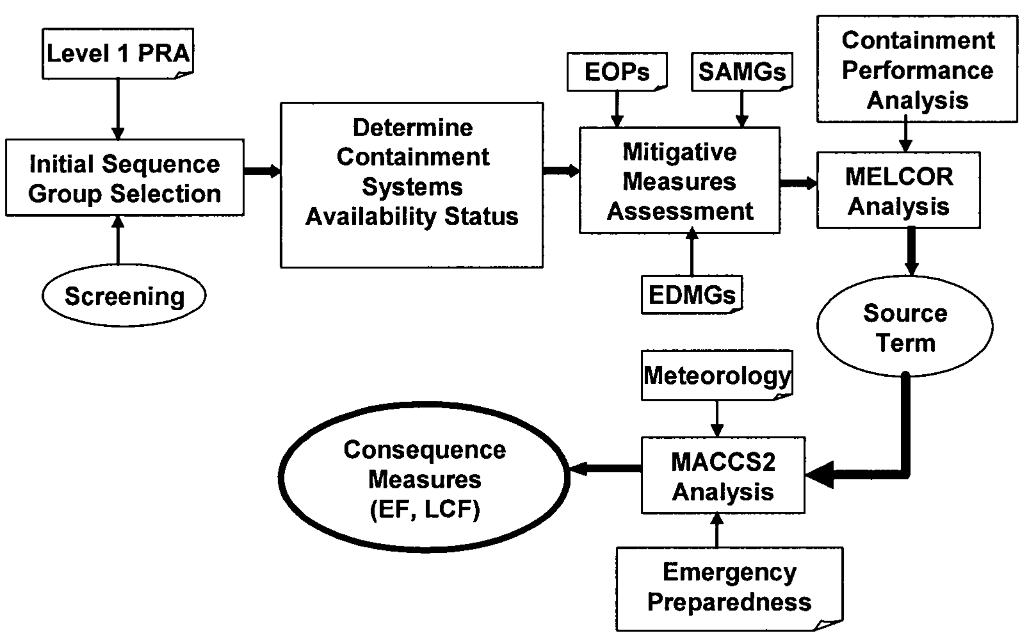

Figure 1 SOARCA Process

The internal events assessment, using SOARCA screening criteria, resulted in retention of no internal event sequences for Peach Bottom, and only containment bypass sequences for Surry. However, for both plants, representative external event sequences were identified.

Screening of the sequences for the subsequent analyses in the SOARCA project was reviewed by an internal peer review panel. One of the concerns of the review panel was that the internal event sequences that would contribute most to risk may not be captured using a strict interpretation of 10^{-6} frequency screening criterion. In the April 14, 2006 SRM, the Commission stated that "in applying a screening radiological release frequency of 10^{-6} per reactor year, the staff should be careful to define release groupings such that release characteristics are representative of scenarios binned into those groups. However, where possible, the groups should also be sufficiently broad to be able to include the potentially risk-significant but lower frequency scenarios (for example, the interfacing systems LOCA scenarios that bypass the containment)". The Review panel indicated that the Commission direction does appear to allow some leeway in this interpretation, and indeed has already been taken in the adaptation of 10^{-7} threshold. The review panel recommended that all Mark I sequence groups leading to reactor vessel breach be considered as potential large early release sequences. The review panel also recommended that all PWR sequences leading to core melt at high RCS pressure be considered as potential large release sequences, given that these sequences could result in a *temperature-induced SGTR. The staff did not agree with the peer review panel* recommendations based on the argument that the SRMs providing the direction for performing this work are unambiguous with respect to the sequence selection criteria and there is no explicit instructions in the SRMs directing the staff to capture sequences that may be risk dominant if they fall below the frequency threshold. It seems that there is a need for a consistent interpretation of "the potentially risk significant but lower frequency scenarios" and a more objective approach to sequence group selection including consideration of uncertainties.

The peer review panel also commented that the screening of the sequences from external events is poorly documented and the basis for the chosen sequences is totally lacking. The review panel noted that it is not clear that there is a basis for treating the seismic, and fire sequences for being equivalent on the basis of their functional characteristics. There is need to separate seismic and fire contributors due to the impact of the former on emergency planning (EP) and possibly on other mitigative measures. The review panel further noted that if the consequence analyses for seismic events are to include consideration of emergency response actions (sheltering and evacuation), it seems appropriate to delineate between low g and high g earthquakes (since EP in high g earthquakes may be ineffective), and to treat emergency response in low g seismic events differently than in internal events (since EP in low g earthquakes may be degraded). Similarly, the feasibility of mitigative measures may be challenged for earthquakes, e.g., use of off-site equipment may be hampered by the state of the roads, bridges, etc. 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 affect the offsite EP functions.

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. In addition, for external event initiated sequences judgments were made on system availability based on the type and severity of the external event initiating event (e.g., large seismic event).

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 assessment of mitigative measures is based on the licensees' EOPs, severe accident mitigating guidelines (SAMGs), and the newly required extreme damage state mitigative guidelines (EDMGs) currently being implemented. Available mitigative measures, including the primary structures, systems, and components (SSCs) (which includes portable equipment), as well as the necessary support systems and resources, were verified to be available based on the initiating event, subsequent failures, and resulting initial conditions. The staff stated that any pre-staged supporting equipment required for successful implementation such as transport capability, fuel, hoses, connectors, tubing, tools, etc. were verified to be available. A time-line of operator actions and equipment lineup or setup times were developed for the implementation of the available mitigative measures using plant procedures, plant conditions, emergency response activities, and time estimates.

The mitigative measures assessment did not use a formal human reliability analysis. However, the staff stated that it did use many of the performance-shaping factors (PSFs) and good practices developed by the NRC as documented in NUREG 1792, entitled "Good Practices for Implementing Human Reliability Analysis (HRA)." It is not clear how the uncertainties associated with the mitigative measures assessment will be addressed.

MELCOR Analysis

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.

The staff has not yet provided any document on accident progression and source term analysis using severe accident code MELCOR. It is not clear how the state-of-knowledge uncertainties associated with the accident progression phenomena and source term analyses are addressed.

The staff has presented some preliminary results of MELCOR analyses during its October 1, 2007, Commission Technical Assistants briefing. It seems that the time intervals between the start of core damage and bottom head failure are predicted to be 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 MELCOE 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 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.

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 being analyzed has been developed based on meteorological data and emergency response parameter.

The staff is evaluating various options for assessment of dose thresholds for latent cancer fatalities. One option is to adopt the official position of the Health Physics Society (HPS) that "recommends against quantitative estimation of health risks below an individual dose of 5 rem in one year or a lifetime dose of 10 rem above that received from natural sources."

The staff has presented some preliminary results of consequence analyses during its October 1, 2007, Commission Technical Assistants briefing. These preliminary results indicate no early fatalities and no latent cancer fatalities. Sensitivity studies, assuming no B5.b mitigative measures, resulted in no early fatalities and latent cancer fatalities of about two order of magnitude lower than reported in the 1992 Siting Study. It should be noted that all these results were obtained by adopting the HPS position for dose thresholds for latent cancer fatalities. The staff has not performed any sensitivity studies with linear non threshold (LNT) assumption.

SUBCOMMITTEE ACTION

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

ADAMS DOCUMENT PROFILE
FOR SUBCOMMITTEE MEETING STATUS REPORT
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