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

May 15, 2012

The Honorable Gregory B. Jaczko
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: STATE-OF-THE-ART REACTOR CONSEQUENCE ANALYSES (SOARCA) PROJECT

Dear Chairman Jaczko:

During the 594th meeting of the Advisory Committee on Reactor Safeguards, May 10-12, 2012, we completed our review of the staff's activities to date regarding the State-of-the-Art Reactor Consequence Analyses (SOARCA) project. Our Subcommittee on Regulatory Policies and Practices also reviewed this matter on June 21, 2010 and April 25, 2012. During these meetings, we had the benefit of discussions with representatives of the NRC staff and their contractors, and of the documents referenced.

RECOMMENDATIONS AND CONCLUSIONS

1. The SOARCA work is a major step forward in developing more realistic, integral deterministic analyses for severe accident progression for selected accident sequences. It can provide a more integrated approach for analyzing important accident sequences in Level 2 and Level 3 probabilistic risk assessments (PRAs), and the insights from these analyses can be useful in the regulatory decision-making process.

2. Priorities for future work related to SOARCA should be the performance of an uncertainty analysis for Surry and the completion of a MACCS2 best practices document.

3. The experience with the Peach Bottom "best estimate" analysis and the associated uncertainty analyses demonstrate that these analyses should be conducted in parallel rather than having the uncertainty analyses be an "add-on" to an already performed "best estimate" analysis.

4. The SOARCA project has provided estimates of the public health consequences of selected scenarios at two plant sites. Although the scenarios considered by SOARCA are generally important contributors to risk, it is not clear what fraction of the risk has been captured without more complete external events PRAs. Comparisons with earlier studies such as NUREG/CR-2239, which were intended to represent the risk from all accident scenarios, should not be made without acknowledging these differences.

5. The selection of parameters, their uncertainty distributions, and their correlations, as well as sensitivity studies to assess the impact of uncertainties that are difficult to quantify, are critical to the Peach Bottom and Surry uncertainty analyses. The uncertainty reports should describe the approaches used to identify the parameters, distributions, and sensitivity studies and justify the bases for omission of parameters or effects of interest not addressed in the uncertainty analyses.

6. Analyses of severe accident progression in a plant with an ice condenser containment would be an important follow-on study. However, such a study should have a lower priority than completion of the ongoing NRC Level 3 PRA study.

BACKGROUND

The SOARCA study considered the Peach Bottom Atomic Power Station in Pennsylvania and the Surry Power Station in Virginia. Peach Bottom is a General Electric boiling water reactor (BWR) design with a Mark I containment. Surry is a Westinghouse pressurized water reactor (PWR) design with a subatmospheric containment. Two major groups of severe accident scenarios were selected for analysis. The first group, common to both Peach Bottom and Surry, includes short term station blackout (SBO) and long term SBO. Both types of SBOs involve a loss of all alternating current (AC) power. The short term SBO also involves the loss of turbine-driven systems due to loss of direct current (DC) control power or loss of the condensate storage tank and therefore proceeds more rapidly to core damage. SBO scenarios can be initiated by external events such as a fire, flood, or earthquake. SOARCA assumes that the SBOs at Peach Bottom and Surry are initiated by a seismic event. The second severe accident scenario group, which was analyzed only for Surry, involves containment bypass. Two containment bypass scenarios were analyzed. The first was a variant of the short term SBO scenario, involving a thermally induced steam generator tube rupture. The second involves an interfacing systems loss-of-coolant accident caused by an unisolated rupture of low head safety injection piping outside containment.

The SOARCA project benefited greatly from the input of a Peer Review Committee that provided an independent review of the analyses. The Peer Review Committee members have extensive expertise in plant design, operation and maintenance, safety and security-related equipment, severe accident phenomenology, emergency preparedness, and radiological health consequences.

DISCUSSION

As stated in NUREG-1935, "State-of-the-Art Reactor Consequence Analyses (SOARCA) Report,"

> *The overall objective of the SOARCA project is to develop a body of knowledge regarding the realistic outcomes of severe reactor accidents. Corresponding and supporting objectives are as follows:*
>
> - *Incorporate the significant plant improvements and changes not reflected in earlier assessments, including system improvements, training and emergency procedures, offsite emergency response, and recent security-related enhancements described in 10 CFR 50.54(hh), as well as plant changes 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 several decades of research into severe accident phenomenology and radiation health effects.*
>
> - *Evaluate the potential benefits of recent security-related mitigation improvements in preventing core damage and reducing or delaying an offsite release, should one occur.*
>
> - *Enable the NRC to communicate severe-accident-related aspects of nuclear safety to stakeholders, including Federal, State, and local authorities; licensees; and the general public.*
>
> - *Update quantification of offsite consequences found in earlier NRC publications, such as NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," issued December 1982.*

The first three objectives have been largely achieved, although the first and third objectives repeat the potential benefits of security related enhancements. The success in achieving the last two objectives is more limited.

Based on the results of severe accident research programs over the past 25 years, several early containment failure modes that were included in earlier studies were excluded from the SOARCA project because of their low likelihood of occurrence. These include alpha mode containment failure caused by an in-vessel steam explosion during melt relocation that simultaneously fails the vessel and the containment; early containment failure caused by direct containment heating in PWR containments; and early containment failure caused by drywell liner melt-through in a wet cavity in Mark I containments.

It has been understood that this improved understanding of containment failure and other work on source terms implied that a consequence analysis like that presented in NUREG/CR-2239 would be overly conservative for most accident scenarios. SOARCA provides a more rigorous quantification of the benefits of this improved understanding, as well as the potential benefits of the Severe Accident Management Guidelines (SAMGs), and the equipment and mitigation strategies associated with 10 CFR 50.54(hh).

The SST1 source term used in NUREG/CR-2239 is significantly larger in magnitude than any of the Peach Bottom source terms calculated in SOARCA, especially for the cesium group. The SST1 source term releases to the atmosphere begin at 1.5 hours. In contrast, for the short term SBO, in which all AC and DC power is lost in the initiating event and which is the most rapidly progressing of the Peach Bottom scenarios, releases begin after ~8 hours. If credit is given for RCIC blackstart,[1] a SAMG measure, releases begin to occur after ~17 hours. The short term SBO accident progresses too rapidly to be mitigated by the portable equipment associated with 10 CFR 50.54(hh). For the long term SBO, in which AC power is lost but DC battery power is available, releases begin after ~20 hours. In this case, there is sufficient time that deployment of portable equipment is feasible. If this deployment is successful, the scenario is mitigated without core damage.

SOARCA provides a comparison of the potential benefits from accident mitigation measures that may be implemented according to the SAMGs and 10 CFR 50.54(hh) strategies. The results from mitigated cases contain full credit for these personnel actions; unmitigated cases contain no credit for these measures. Human reliability analyses were not performed to quantify the probabilities for success or failure of these mitigation actions. Scenarios that contribute to both the mitigated and unmitigated results also contain assumptions about preceding successes and failures of selected operator actions that are contained in the plant-specific Emergency Operating Procedures.

The MACCS2 consequence code used in SOARCA also has a number of improvements over earlier codes. These include capability to describe wind directions in 64 compass directions (instead of 16) and up to 20 emergency-phase cohorts (instead of the original limit of 3) to describe variations in emergency response by segments of the population (SOARCA used six cohorts in the analysis). Site specific evacuation time estimates (ETEs) provided by the licensees were used to develop speeds for evacuating cohorts. A limited and conservative seismic analysis of local infrastructure, which may affect evacuation activities for each site, was performed. The seismic analysis indicated that long-span bridges would be unlikely to survive and were assumed impassable. Some smaller bridges and road crossings, as well as some roadways where underlying soils could slide off into adjacent waterways, were also assumed to have failed. Response parameters that may be affected by an earthquake (e.g., mobilization of the public, evacuation speed, shielding) were adjusted to reflect the potential impact.

[1] Blackstart of the reactor core isolation cooling (RCIC) system refers to starting RCIC without AC or DC control power.

With this more realistic simulation of emergency response, accident progression, source term analysis, and dispersion; SOARCA predicts that there is essentially no risk of early fatalities, even for the unmitigated scenarios, as close-in populations were evacuated before or shortly after plume arrival.

Latent cancer fatality health effects were assessed using a linear no-threshold; a 620 mrem threshold; and the Health Physics Society 5 rem/year, 10 rem lifetime thresholds. For the scenarios considered, latent health effects calculated using any of these dose-response models (in combination with the frequency of release) are small in comparison to the NRC Safety Goal. Most of the latent cancer fatality risk is associated with the small doses received by populations returning to their homes in accordance with the emergency planning guidelines established by the affected state authority.

The analyses in the SOARCA project are described as "best-estimate" analyses. NUREG-1935 does not define "best estimate." During discussions with the staff and contractors, it appeared that the consensus was that "best estimate" is intended to mean "most likely." While it is important to have insights into the "most likely" outcome of a scenario, for situations where the uncertainties are large, it is critical to understand the impact that such uncertainties may have on the outcomes. As a part of the SOARCA project, a number of uncertainty and sensitivity studies, including several suggested by the Peer Review Committee, have been performed. These are still in progress for Peach Bottom. It appears that some of the ongoing analyses could affect the "best-estimate" results described in the current NUREG and NUREG/CR reports issued for comment.

The selection of parameters, their uncertainty distributions, and their correlations, as well as sensitivity studies to assess the impact of uncertainties that are difficult to quantify, are critical to the uncertainty analysis. The approach used to identify these parameters is not clear. In Section 2.2 of the draft NUREG/CR report on uncertainty analysis, it is stated that "the approach is based on a formalized PIRT (phenomena identification, and ranking table) process," while in Section 4 it is stated that the "uncertain parameters and their distributions were identified/characterized through an informal elicitation of subject matter experts." The report should be consistent and accurate in its description of the approach used to identify the parameters and distributions and the selection of sensitivity studies.

The categorization of the parameters into groups such as sequence issues, in-vessel progression, ex-vessel progression, etc. is helpful and can aid subject matter experts in evaluating whether the selected parameters encompass the phenomena of interest. Table 2.2.1 of the draft NUREG/CR report on uncertainty analysis identifies the groups and parameters. It would be helpful to have a comparable summary table in Section 4 of that report to identify the method used to identify the distribution such as "expert judgment", "test results", "additional analysis", etc. The uncertainty reports should also describe the approaches used to identify the parameters, distributions, and sensitivity studies and justify the bases for omission of parameters or effects of interest not addressed in the uncertainty analysis.

There are a number of physical processes involved during in-vessel and ex-vessel accident progression where uncertainties are large. Not all of these uncertainties can be represented by parameter distributions. One way to address such phenomenological uncertainties is sensitivity analyses for alternative models. For example, one area of uncertainty identified by the Peer Review Committee that the staff may have not explored sufficiently is lower head failure. MELCOR focuses on creep rupture as the dominant mechanism for lower head failure. The staff argues that this approach is supported by experimental results in NUREG/CR-5582, which were performed on one-fifth scale models representing PWR heads. They also argue that the timing differences between gross lower head failure and penetration failure with the available penetration model are not significant to the overall accident progression. The staff also notes that the penetration model is a simple lumped parameter model for bulk heat transfer and is not adequate to calculate molten material drainage into a BWR reactor pressure vessel drain line. More detailed analyses in NUREG/CR-5642 suggest that in certain scenarios failure of this penetration is more important than failure of other lower head penetrations or the vessel. Sensitivity studies could help to evaluate whether earlier failure of the drain line would have a significant impact on outcomes.

The SOARCA project has provided estimates of the public health consequences of selected scenarios at two plant sites. Modern Level 1 internal events PRAs were available for Peach Bottom and Surry. No modern seismic PRAs were available, although seismic PRAs were done for both plants as part of NUREG-1150 and for Peach Bottom as part of the Individual Plant Examination of External Events (IPEEE). As a result, the SOARCA external event scenarios and the estimated frequencies of these scenarios were based on expert judgment that considered the impact of changes in the seismic hazard and methods on the published external event PRA results.

Although station blackout sequences are generally important contributors to risk, without more complete external events PRAs, it is not clear what fraction of the risk has been captured even for these two plants.

Comparisons with earlier studies such as NUREG/CR-2239, which were intended to represent the risk from all accident scenarios, should not be made without acknowledging limitations in the analyses. For the most part the documentation does this. However, this is not always the case. For example, the statement in the brochure that “SOARCA does not examine all scenarios typically considered in PRA, even though it includes the important scenarios” overstates the completeness of the analyses for Surry and Peach Bottom. The staff in its revision of the SOARCA documentation should be careful to avoid such overstatement.

A great deal of important technical work has been accomplished as part of the SOARCA project, and this work should be documented. A draft summary NUREG has been prepared, and more detailed NUREG reports that describe the analyses at Peach Bottom and Surry have been issued. A best practices document has also been prepared for MELCOR. These documents appear to have been developed in some haste, but the staff has told us that revisions are planned. The uncertainty analysis of Peach Bottom is in progress, and it appears a best practices document for MACCS2 is being developed. These should be completed. The highest priority for future work related to SOARCA should be the performance of an uncertainty analysis for Surry.

Analyses of severe accident progression in a plant with an ice condenser containment would be an important follow-on study for this important containment type. However, such a study has a lower priority than completion of the NRC sponsored Level 3 PRA study.

We commend the staff on its efforts in performing the consequence analyses for Peach Bottom and Surry. The SOARCA work is a major step forward in developing more realistic, integral deterministic analyses for severe accident progression for selected accident sequences. It can provide a new more integrated approach for analyzing important accident sequences in Level 2 and Level 3 PRAs, and the insights from these analyses can be useful in the regulatory decision-making process.

Dr. Dana A. Powers did not participate in the Committee's deliberations regarding this matter.

Sincerely,

/RA/

J. S. Armijo

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Letter to The Honorable Gregory B. Jaczko, NRC Chairman, from J. Sam Armijo, ACRS Chairman, dated May 15, 2012

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