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

February 25, 2008

The Honorable Dale E. Klein
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

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

Dear Chairman Klein:

During the 549th meeting of the Advisory Committee on Reactor Safeguards, February 7-9, 2008, we completed our review of the staff's activities to date regarding the State-of-the-Art Reactor Consequence Analyses (SOARCA) Project. We had discussed this matter previously during our meetings on September 7-9, December 7-9, 2006, and December 6-8, 2007. Our Subcommittee on Regulatory Policies and Practices also reviewed this matter on July 10 and November 16, 2007. During these meetings, we had the benefit of discussions with representatives of the NRC staff and of the documents referenced. We also heard the remarks by a representative of the Union of Concerned Scientists regarding the SOARCA project during our meeting on December 6-8, 2007.

RECOMMENDATIONS

1. Level-3 probabilistic risk assessments (PRAs) should be performed for the pilot plants before extending the analyses to other plants. The PRAs should address the impact of mitigative measures using realistic evaluations of accident progression and offsite consequences. The core damage frequency (CDF) should not be the basis for screening accident sequences.

2. 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.

3. Consequences should 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.

DISCUSSION

The staff is currently implementing its plan for developing state-of-the-art reactor consequence analyses. This work will: (1) evaluate and update, as appropriate, analytical methods and models for realistic evaluation of severe accident progression and offsite consequences; (2) develop state-of-the-art reactor consequence assessments of severe accidents; and

(3) identify mitigative measures that have the potential to significantly reduce risk or offsite consequences. The analyses include external events; consideration of all mitigative measures, including the newly required extreme damage state mitigative guidelines (B.5.b); state-of-the-art accident progression modeling based on 25 years of research to provide a best estimate for accident progression, containment performance, time of release, and fission product behavior; more realistic offsite dispersion modeling; and site-specific evaluation of public evacuation based on updated emergency plans.

In a Staff Requirements Memorandum dated April 14, 2006, the Commission stated that the staff's proposal to examine significant radiological release scenarios having estimated likelihoods of one in a million or greater per year is an appropriate initial focus. Because a significant radiological release cannot occur without core damage and because the current understanding of Level-1 events is more complete than the subsequent progression, the screening was done on the basis of a CDF greater than or equal to 1×10^{-6} per reactor year. For bypass events, a lower screening frequency is used, a CDF greater than or equal to 1×10^{-7} per reactor year. Because not all CDF events will lead to significant radiological releases, this screening approach is somewhat more inclusive than the initial staff proposal. Sequences are grouped based on functional characteristics, and the frequency of the group is used as the basis for comparison with the screening criteria.

Experience from contemporary full-scope PRAs demonstrates that there are problems associated with the use of CDF as a numerical screening criterion to restrict the scope of subsequent Level-2 and Level-3 analyses. In such PRAs, the most important contributors to offsite consequences are not necessarily significant contributors to CDF, and are not necessarily characterized by initial containment bypass events. The number of these sequences and their aggregate contribution to overall plant risk can increase dramatically as the numerical cutoff is reduced. Thus, application of *a priori* CDF screening criteria can inappropriately overlook many risk-significant scenarios. Such an approach also does not provide a fully integrated evaluation of risk in terms of frequency and consequences.

With current computational capabilities, virtually all sequences can be considered through the complete Level-1, Level-2, and Level-3 analyses. Uncertainties at each stage of the process can also be propagated through the full accident scenarios. This type of fully integrated evaluation removes the need for intermediate screening and scenario grouping. It allows for clear identification of the most important scenarios for offsite consequences and facilitates an integrated evaluation of important physical and functional dependencies that affect core damage, severe accident progression, and offsite emergency responses.

The staff argues that events below the current cutoff frequency can become highly uncertain. Although it is true that the uncertainties associated with less frequent scenarios generally increase, it is important to be aware of the potential for severe consequences in regulatory decisionmaking and in assessing defense-in-depth requirements.

One of the arguments for the SOARCA program is the need to update and replace the site-specific quantification of offsite consequences found in NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," (issued 1982), and NUREG/CR-2723, "Estimates of the Financial Consequences of Nuclear Power Reactor Accidents," (issued 1982). It has long been recognized that results of these studies are overly conservative and that the most realistic assessments are those in NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," (issued 1990), and related studies such as NUREG/CR-6295, "Reassessment of Selected Factors Affecting Siting of Nuclear Power Plants," (issued 1997).

However, NUREG-1150 is based on state of knowledge and understanding of severe accidents from the 1980s. As we now envision a future in which current reactors will be operating for an additional 20-40 years and new reactors will be built, it is timely to consider updating our understanding of the risks of nuclear power.

Level-3 PRAs for internal and external events based on current PRA and severe accident technology, updated plant configurations and mitigative measures such as emergency operating procedures (EOPs), severe accident management guidelines (SAMGs), and the newly required extreme damage state mitigative guidelines (B.5.b) should be performed. Such PRAs would require a substantially greater commitment of resources than SOARCA. However, as a minimum, a limited set of updated Level-3 PRAs for the SOARCA pilot plants should 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. 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. In addition, the Level-3 PRAs would update both the technology and results of NUREG-1150.

Like SOARCA, the proposed PRAs should consider at-power conditions. The intent is to primarily use existing technology and knowledge. Because additional research is required to better understand and characterize the shutdown source term, the at-power Level-3 PRAs should be completed before addressing risk at shutdown.

For internal events, the application of the SOARCA process to the pilot plants seems scrutable. The sequence groups examined represent more than 90% of the total CDF. The process for selecting sequences for external events is less clear. The process is intended to draw upon external event (EE) sequences determined using available plant-specific data and assessments (e.g. NUREG-1150), SPAR-EE (Standardized Plant Analysis Risk-External Event) model information, and generic insights from available literature. However, no comparisons have been presented between the seismic event sequences chosen for Surry and Peach Bottom and those reported in NUREG/CR-4550, and no estimate of the fraction of the external event CDF covered by the sequences considered has been presented. The selection seems more motivated by generic insights. More importantly, unlike in the seismic studies supporting the NUREG-1150 study reported in NUREG/CR-4550, no association of the frequency of the sequence with the peak ground acceleration of the earthquake is provided. Such an association may be important in assessing the effectiveness of emergency planning in dealing with the consequences of a seismically induced event. Since the results of the pilot studies indicate that external event sequences are the most significant in terms of consequences to the public, a more complete and detailed examination of these events appears warranted.

The staff is planning to address the impacts of seismic events on emergency planning through sensitivity studies. Because of the risk significance of a large seismic event, it is important that an estimate of the impacts of the event on emergency planning response be made as realistic as feasible to anchor the sensitivity studies.

In either a consequence analysis or a Level-3 PRA, a critical element in calculating the consequences is the choice of a model for the calculation of latent cancer fatalities. Previous NRC studies have used the LNT model. Among other options, the staff is evaluating use of a threshold based on the Health Physics Society Position Statement (5 rem in a year or 10 rem in a lifetime). This Position Statement indicates that below such dose levels, estimates of risk should only be qualitative, i.e, expressed as a range based on the uncertainties in estimating risk, emphasizing the inability to detect any increased health detriment. However, this Statement does not provide any guidance on how to estimate the range of consequences below this level. Other authorities such as the National Academy of Sciences, the World Health Organization, and the National Council on Radiation Protection and Measurement still support use of the LNT model.

It seems clear that the health detriments at radiation levels below 5 rem are so small that they cannot be detected by epidemiological studies. Until a much greater understanding of cell damage and repair mechanisms is achieved, the actual existence of a threshold can be neither proved nor disproved. However, as a practical matter, we see no way to estimate the range of consequences below this level except by using the 5 rem threshold and some lower threshold to perform the consequence calculations. This does not necessarily imply the use of a zero rem lower threshold. For rare events such as a serious nuclear reactor accident, consequences comparable to those resulting from a typical yearly exposure to natural radiation, i.e., 300 mrem, could be deemed not to represent an undue risk. A calculation with a zero rem threshold should be included for comparison with historical results. Even in this case, a de facto threshold is introduced, because the transport calculations become meaningless at large distances and the calculation must be truncated at some distance.

We commend the staff on its efforts in performing the consequence analyses for Peach Bottom and Surry. We look forward to further interactions with the staff as the study proceeds.

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

Sincerely,

/RA/

William J. Shack
Chairman

References:

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2. Memorandum to Luis A. Reyes, Executive Director for Operations, NRC, from Kenneth R. Hart, Acting Secretary, NRC; Subject: Staff Requirements – SECY-05-0233- Plan for Developing State-of-the-Art Reactor Consequence Analyses, dated April, 14, 2006. (Official Use Only - Sensitive Internal Information, Limited to NRC Unless the Commission Determines Otherwise)

3. U.S. Nuclear Regulatory Commission, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, Vol. 1, December 1990.

4. U.S. Nuclear Regulatory Commission, “Analysis of Core Damage Frequency: Surry Power Station, Unit 1, External Events,” Sandia National Laboratories, NUREG/CR-4550, Vol. 3, Rev. 1, Part 3, December 1990.

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6. U.S. Nuclear Regulatory Commission, “Technical Guidance for Siting Criteria Development,” Sandia National Laboratories, NUREG/CR-2239, December 1982.

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8. U.S. Nuclear Regulatory Commission, “Reassessment of Selected Factors Affecting Siting of Nuclear Power Plants,” Brookhaven National Laboratories, NUREG/CR-6295, February 1997.

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