# POLICY ISSUE
## NOTATION VOTE

July 5, 2012 SECY-12-0092

FOR: The Commissioners

FROM: R. W. Borchardt
Executive Director for Operations

SUBJECT: STATE-OF-THE-ART REACTOR CONSEQUENCE ANALYSES - RECOMMENDATION FOR LIMITED ADDITIONAL ANALYSIS

PURPOSE:

The purpose of this memorandum is to respond to Staff Requirements Memorandum (SRM) SRM-SECY-05-0233, "Plan for Developing State-of-the-Art Reactor Consequence Analyses," dated April 14, 2006 (WITS 200600236/200600237), and SRM-COMSECY-06-0064, "State-of-the-Art Reactor Consequence Analyses Communications Plan," dated April 2, 2007 (WITS 200700117/200800373). Specifically, the Commission asked the staff to provide a recommendation based on initial studies as to whether continuing the State-of-the-Art Reactor Consequence Analyses (SOARCA) project as originally described in SECY-05-0233, "Plan for Developing State-of-the-Art Reactor Consequence Analyses," dated December 22, 2005, is necessary to achieve its objectives. The staff is seeking Commission approval to perform limited additional analyses to meet the objectives of SOARCA.

SUMMARY:

The staff has completed the SOARCA project for two pilot plants, the Peach Bottom Atomic Power Station (Peach Bottom), which is a boiling water reactor (BWR) with a Mark I containment and the Surry Power Station (Surry), which is a pressurized water reactor (PWR) with a large dry (subatmospheric) containment. In order to provide the Commission a recommendation on whether and how to continue the SOARCA project, the staff sought input from a number of sources. Based on its review, the staff recommends performing limited severe accident consequence analyses for a PWR plant with an ice condenser containment

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because these analyses will provide additional knowledge of severe accident progression and consequences in a plant with a unique containment design. The results of this analysis could inform agency activities such as the implementation of the Near Term Task Force (NTTF) recommendations related to hydrogen control, reliable hardened vents, or filtered vents for reactor designs other than BWRs with Mark I and Mark II containments. The staff also recommends performing an uncertainty analysis (UA) for a PWR station blackout (SBO) scenario. The ongoing UA for a Peach Bottom scenario, expected to be completed in late 2012, is providing important insights regarding how uncertainties in the most influential parameters associated with severe accident progression and consequences in a BWR affect the results. A UA for a Surry SBO scenario would be expected to provide similar types of insights for PWR severe accident progression and this would inform the Site Level 3 Probabilistic Risk Assessment (PRA) project for the Vogtle Electric Generating Plant (Vogtle) Units 1 and 2, which are also PWRs.

DISCUSSION:

The staff provided the Commission a memorandum titled, “State-of-the-Art Reactor Consequence Analyses – Peach Bottom and Surry Results” (ML12132A213), dated June 22, 2012, which enclosed the completed SOARCA analyses of two pilot plants, Peach Bottom and Surry. The objectives of the SOARCA project as described in SECY-05-0233 are the following:

- Using a methodology based on state-of-the-art analytical tools, determine best estimates of the radiological dose consequences (including early and latent fatalities and land contamination) for each U.S. operating reactor site and present those results using risk communication techniques; and
- Achieve informed public understanding of the following factors:
  - Extent and value of defense-in-depth features of plant design and operation, including mitigative strategies that are employed to reduce risk
  - Most significant influential assumptions.

The staff believes that the analyses of the important scenarios for the two pilot plants have generally met the objectives stated above. The NUREG reports (NUREG-1935, “State-of-the-Art Reactor Consequence Analyses Report,” NUREG/CR-7110 Volume 1, “Peach Bottom Integrated Analysis,” and Volume 2, “Surry Integrated Analysis,” and NUREG/BR-0359, “Modeling Potential Reactor Accident Consequences”) that staff recently completed on the SOARCA project document the following: (1) the state-of-the-art analytical models and methods used in the SOARCA project, (2) the best estimate radiological dose consequences, including early and latent fatality risks, for important scenarios for the Peach Bottom and Surry plants, (3) the results presented using risk communication techniques, for example through a public information brochure and public meetings, and (4) the most influential assumptions such as failure criteria for BWR safety relief valves. The UA of a Peach Bottom SBO scenario currently in progress will add to staff and public understanding of the most significant influential assumptions in the SOARCA project. The Peach Bottom UA will be completed in late 2012, as well as the MELCOR and MELCOR Accident Consequence Code System, Version 2 (MACCS2) Best Practices NUREG/CRs.

Regarding public understanding of defense-in-depth features including mitigative strategies, SOARCA modeled mitigation measures including those in emergency operating procedures (EOPs), severe accident management guidelines (SAMGs), and Title 10 to the *Code of Federal Regulations* (10 CFR) 50.54(hh) equipment and strategies required following the terrorist attacks of September 11, 2001. To assess the benefits of 10 CFR 50.54(hh) mitigation measures and to provide a basis for comparison to the past analyses of unmitigated severe accident scenarios, the SOARCA project also analyzed each scenario without 10 CFR 50.54(hh) equipment and procedures. The analysis that credits successful implementation of the 10 CFR 50.54(hh) equipment and procedures in addition to actions directed by the EOPs and SAMGs is referred to as the mitigated case. The analysis without 10 CFR 50.54(hh) equipment and procedures is referred to as the unmitigated case. The unmitigated case of the Surry interfacing systems loss-of-coolant accident is an exception to this general principle because it was necessary to assume that at least one of the EOP actions failed to occur for the scenario to lead to core damage and radioactive release. Chapter 3 of NUREG/CR-7110, Volumes 1 and 2 details the specific equipment and operator actions credited for each scenario. In addition, the SOARCA public information brochure (NUREG/BR-0359) includes a discussion of mitigation and the results of the mitigated and unmitigated scenarios.

SOARCA analyses indicate that successful implementation of existing mitigation measures can prevent reactor core damage or delay or reduce offsite releases of radioactive material. All SOARCA scenarios, even when unmitigated, progress more slowly and release much less radioactive material than in earlier studies (e.g., NUREG/CR-2239, “Technical Guidance for Siting Criteria Development,” commonly referred to as the 1982 Siting Study). The SOARCA analyses show that emergency response programs implemented as planned and practiced, would reduce the scenario-specific risk of health consequences among the public during a severe reactor accident at the plants evaluated. As a result, the calculated risks of public health consequences from severe accidents modeled in SOARCA are very small.

The SOARCA project’s scope initially included analysis of all operating reactors; however SRM-COMSECY-06-0064 reduced the scope to a subset of eight plants with first assessments of one BWR and one PWR. The staff has concluded it is not necessary to analyze all plant sites or even all plant types to achieve the objectives of the SOARCA project. The analyses for Peach Bottom and Surry updated the understanding of severe accident progression, mitigation, and consequences for two major plant designs—the BWR with a Mark I containment and the PWR with a large dry (subatmospheric) containment. These two designs encompass 80 percent of the designs of the current operating fleet (83 of 104 operating reactors).

In order to provide the Commission a recommendation on whether and how to continue the SOARCA project, the staff sought input from a number of sources. These sources include the historical record of staff interactions with the Commission, the external SOARCA peer review committee, the internal SOARCA steering committee, the Advisory Committee on Reactor Safeguards (ACRS), and members of the public. Further, staff also used Fukushima lessons learned to identify potential open issues that would necessitate additional research.

Based on this review, the staff identified the following items for potential further work or to address open issues raised during the pilot plant studies. To avoid duplication and ensure efficient use of agency resources, the staff evaluated whether the items require NRC action, are being addressed by other agency activities, or whether follow-on SOARCA-related research would be the best alternative to address the item. For each of the items except the last two, the

staff determined that either other agency activities (noted in parentheses) are being conducted or planned to address the issue, the item is no longer needed, or appropriate information is not currently available to analyze the item. The following paragraphs provide details on the staff's disposition of these items.

- Spent fuel pool risk (Fukushima Tier 3 spent fuel transfer recommendation, Site Level 3 PRA)
- Multi-unit and site risk (Site Level 3 PRA)
- Low power, shutdown, and other modes (Site Level 3 PRA)
- Human reliability analysis (Site Level 3 PRA)
- Economic consequences and land contamination (Commission paper under development)
- Additional comparisons to Fukushima (DOE/NRC Forensic Analysis)
- Large seismic event and soil liquefaction (Site Level 3 PRA and Seismic research plan)
- Development of an integrated, predictive computer-based tool (no longer needed)
- Cancer incidence rates (requires updated EPA risk coefficients)
- Other plant and containment types – e.g., Ice condenser, Mark II, III, CE, B&W
- UA for a Surry scenario

Many stakeholders hold the view that the spent fuel pools, multiple units, and operating modes other than full power should be analyzed. Stakeholders also feel that a human reliability analysis for mitigation measures would be informative. SOARCA was not intended to be the equivalent of a full-scope level 3 PRA. Rather, it provides a state-of-the-art body of knowledge, models, and methodologies about severe accident consequences in two representative reactor designs that can be applied as needed to agency activities. The ongoing Site Level 3 PRA of the Vogle Units 1 & 2, which will analyze spent fuel pool risk, multi-unit and site risk, all operating modes, and human reliability, will further add to this body of knowledge, models, and methodologies. The Site Level 3 PRA plans to use the updated modeling and methods that were developed for SOARCA and is scheduled for completion in 2016. Additionally, SOARCA models and methods will inform the staff's Tier 3 lessons-learned activity related to the potential transfer of spent fuel from pools to casks.

As described in SECY-05-0233, the SOARCA project's original scope included calculating offsite consequences of severe accidents in terms of (1) health effects and (2) land contamination. However, in SRM-COMPBL-08-002/COMGBJ-08-0003, "Economic Consequence Model" dated September 10, 2008, the Commission directed the staff to complete the SOARCA project to assess offsite health consequences and to not delay the project in order to include an assessment of economic consequences and land contamination.

The Fukushima accident occurred when SOARCA was nearing completion. Because the Peach Bottom plant studied in the SOARCA project has a similar design to the Fukushima plants, stakeholders posed many questions on how SOARCA Peach Bottom calculations compared with actual Fukushima outcomes. The SOARCA report (NUREG-1935) includes a qualitative discussion comparing the Peach Bottom analysis to Fukushima in five areas—(1) operation of the reactor core isolation cooling system, (2) hydrogen release and combustion, (3) 48-hour truncation of releases in SOARCA, (4) multi-unit risk, and (5) spent fuel pool risk. Additionally, Sandia National Laboratories is conducting a joint NRC and U.S. Department of Energy (DOE) Fukushima Forensic Analysis targeted for completion later this year. This analysis should provide additional insights into the Fukushima accident. It is also expected that international

cooperative efforts sponsored by the Organization for Economic Cooperation and Development's Nuclear Energy Agency and the International Atomic Energy Agency over the next few years will provide ongoing analysis of Fukushima with NRC and other countries' codes as information becomes available.

The SOARCA study analyzed SBOs assumed to be initiated by low probability seismic events. SOARCA excluded more extreme seismic events of even lower probability which could conceivably cause immediate containment failure followed by core damage. Seismic fragility quantification for these extreme and rare seismic events, in particular quantification of the size of a hole or amount of leakage, is currently subject to considerable uncertainty. More research is needed before undertaking a realistic, best-estimate analysis of such rare events. Soil liquefaction analysis will be included in the Site Level 3 PRA project.

In SECY-05-0233, the staff proposed developing an integrated, predictive, computer-based tool to assist decisionmaking in the event of a severe reactor accident. Staff envisioned that this tool could model and predict accident progression faster than real time and thereby aid the NRC's incident response center in the event of a severe reactor accident. Resources needed to develop this tool were redirected to higher priority work including completion of the SOARCA pilot plant analyses for Peach Bottom and Surry. Increases in computing capability allow existing codes to perform analyses much faster than in the past. As a result of the slow progression of the Fukushima accident and our demonstrated ability to use existing tools including MELCOR, MACCS2, and the Radiological Assessment System for Consequence Analysis (RASCAL) code during the Fukushima response, the staff concludes that the development of another predictive computer-based tool is not necessary. However, staff is evaluating enhancements to existing tools based on the Fukushima experience, and they would be deployed as necessary in the agency's incident response program.

Some stakeholders noted that a state-of-the-art report should report cancer incidence as well as, or instead of, cancer mortality since that is state-of-the-art in epidemiology. The SOARCA project reported offsite consequences in terms of individual average latent cancer fatality risk to provide appropriate context of the risk in relation to other risks and to enable comparison to the NRC Safety Goal. SOARCA used latent cancer expression coefficients for the United States population based on Biological Effects of Ionizing Radiation (BEIR) V risk projection models, as detailed in the U.S. Environmental Protection Agency's (EPA's) publication "Estimating Radiogenic Cancer Risks" (EPA 402-R-93-076, 1994) and implemented in the EPA's Federal Guidance Report 13, "Cancer Risk Coefficients for Environmental Exposure to Radionuclides" (FGR-13). The BEIR V report used cancer mortality as a metric because, at that time, most epidemiological studies were based on cancer mortality, not cancer incidence. The EPA has not yet incorporated cancer incidence data from the BEIR VII report into a revision of FGR-13, so there are no new updated cancer risk coefficients currently available.

SECY-05-0233 envisioned applying the SOARCA methodology to all operating reactors. Subsequently, SRM-COMSECY-06-0064 directed the staff to analyze just eight reactors representing the different plant/containment types, starting with one BWR and one PWR. Three plants had volunteered for the project—Peach Bottom, Surry, and the Sequoyah Nuclear Plant (Sequoyah). The SOARCA team started analysis of these three plants but then stopped its analysis of Sequoyah in order to focus resources on the Peach Bottom and Surry analyses. The staff evaluated the SOARCA pilot plant results and the improved models and methods for

severe accident and consequence analysis for each of the remaining six plant/containment designs to determine whether further analyses are necessary to meet the project's objectives. Due to the similarities of the other six designs to the designs of Peach Bottom and Surry, staff believes that these analyses would provide limited additional insight and thus would not be cost effective, as discussed below. However, a severe accident consequence analysis of SBOs at a PWR with an ice condenser containment would inform ongoing agency activities and be cost effective since staff has already initiated the analysis of Sequoyah.

The ice condenser containment is unique relative to other containment types including those studied in the SOARCA project (BWR Mark I at Peach Bottom and large dry at Surry) and that to be studied in the Site Level 3 PRA project (large dry at Vogtle 1 & 2). It has aspects worth exploring regarding buildup of hydrogen during an SBO since it has the lowest design pressure among containments in the United States and relies on hydrogen igniters. Sequoyah provided plant information for the original SOARCA analyses, and staff would continue the analysis with some resource savings. The NRC currently has a partially developed MELCOR model of Sequoyah. Further analysis would be limited to a seismically induced SBO event since the SOARCA project, Fukushima, and other studies have shown SBOs to be the most important scenarios.

The BWR with a Mark III containment design has similar aspects regarding hydrogen during an SBO as the PWR with an ice condenser containment. However, staff does not recommend analyzing the BWR Mark III. The BWR Mark III has a larger containment volume and greater containment design pressure than the ice condenser containment, so the ice condenser consequence analysis is considered more likely to reveal a more extreme accident progression. Also, the BWR Mark III design is less prevalent than the PWR ice condenser design (4 BWR Mark III units vs. 9 PWR ice condenser units currently operating[1]).

The staff does not recommend conducting severe accident consequences for a BWR Mark II design either. The BWR Mark II design is similar in concept to the Mark I design but with a simplified suppression pool design and a more unified containment structure. Containment heat removal systems (sprays and suppression pool cooling) and nitrogen inerting strategies are the same as for the Mark I containments. Containment venting can also be performed in a similar fashion to the Mark I containments. Due to these similarities, the staff believes that its analysis of Peach Bottom bounds the accident progression and consequences expected for an SBO at a BWR Mark II plant and therefore an analysis of Mark II plant would not be cost effective.

The staff also believes that severe accident consequence analyses of other PWR designs with large dry containments beyond Surry would not be cost effective. Although Surry has a subatmospheric large dry containment, the difference between its containment and a standard large dry containment is not expected to impact severe accident progression and offsite consequences. The major difference is that the containment is maintained at a negative pressure with respect to the outside atmosphere. This negative pressure provides some additional margin for response to design basis accidents but is insignificant for severe

[1] Currently operating PWR plants with an ice condenser containment: Catawba Nuclear Station Units 1 and 2, Donald C. Cook Nuclear Power Plant Units 1 and 2, McGuire Nuclear Station Units 1 and 2, Sequoyah Nuclear Plant Units 1 and 2, and Watts Bar Nuclear Plant Unit 1

Currently operating BWR Mark III plants: Clinton Power Station Unit 1, Grand Gulf Nuclear Station Unit 1, Perry Nuclear Power Plant Unit 1, and River Bend Station Unit 1

accidents. The staff also recommends against analyzing a Babcock and Wilcox (B&W) designed PWR with a large dry containment or a Combustion Engineering (CE) designed PWR with a large dry containment. While the B&W and CE nuclear steam supply system (NSSS) designs are different than the Westinghouse NSSS design in the Surry plant, the differences are mostly in the design and size of the steam generators. A severe accident consequence analysis of B&W and CE designs could reveal some limited insights regarding accident progression, but the staff believes these would not be cost effective.

Beyond the goal of more completely meeting the SOARCA project's original objectives, the consequence analysis of an ice condenser plant would provide insights and results to inform agency activities such as responses to NTTF recommendations 5.1 (reliable hardened vents for Mark I and Mark II containments) and 5.2 (reliable hardened vents for containment designs other than Mark I and II), 6 (hydrogen control and mitigation inside containment or in other buildings), and 12.2 (enhance training on severe accidents).

The SOARCA project included within its scope a UA for a severe accident scenario at one of the two pilot plants studied. The unmitigated long-term SBO at Peach Bottom was selected because its release timing and consequences are characteristic of the majority of unmitigated scenarios analyzed in SOARCA. Since this was a first-of-a-kind study, significant resources were required to develop the methodology to conduct the UA. Based on progress to date, the Peach Bottom UA has revealed many important insights about accident progression in a BWR with a Mark I containment. For example, the Peach Bottom UA has shown that the failure criteria for safety relief valves (SRV) is critical to determine whether the reactor coolant system fails by main steam line creep rupture or by SRV failure. A UA for a Surry scenario would be expected to reveal similar insights for PWR severe accident progression and these insights would directly support the Site Level 3 PRA project. For example, the Surry UA could provide data regarding reactor coolant system and containment failure modes and subsequent releases that could support the choices for split fractions and release fractions in the Level 2 component of the PRA.

Pending available resources, the staff considers the ice condenser plant analysis higher priority than the Surry UA. This is because the ice condenser analysis will provide additional knowledge of severe accident progression in another containment design.
In the May 15, 2012 letter (ML12135A385) from J. S. Armijo to Chairman Jaczko, the ACRS commended the staff's analyses of Peach Bottom and Surry and endorsed a follow-on study to analyze severe accident progression in a plant with an ice condenser containment, noting that it should be a lower priority than completion of the Site Level 3 PRA project. Staff would coordinate the Sequoyah analysis and Surry UA with the Site Level 3 PRA project as well as other activities supporting the Fukushima lessons learned such that schedules for all activities are not adversely impacted. The ACRS also recommended staff perform a UA for Surry. Staff anticipates performing a UA for selected Surry parameters that would inform the Site Level 3 PRA as the Vogtle MELCOR model is being completed. These analyses would also add to the body of knowledge on severe accident progression and consequences developed in SOARCA.

RECOMMENDATION:

The staff recommends that the Commission approve limited continuation of the SOARCA project. Based on the review of open items related to severe accident research, the staff recommends the following two analyses:

(1) Severe accident consequence analysis of an SBO at a PWR with an ice condenser containment (Sequoyah) based on the insights from the Peach Bottom and Surry analyses.
(2) UA for a severe accident scenario at Surry.

RESOURCES:

Resources proposed to complete the activities discussed above are described in the non-public Enclosure.

COORDINATION:

The Office of the Chief Financial Officer has reviewed this paper for resource implications and has no objections. The Office of the General Counsel has reviewed this paper and has no legal objection.

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R. W. Borchardt
Executive Director
for Operations

Enclosure:
Projected Resource Needs for SOARCA Recommendations for Limited Additional Analyses

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Enclosure:
Projected Resource Needs for SOARCA Recommendations for Limited Additional Analyses

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