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State-of-the-Art Reactor Consequence Analyses (SOARCA) Report

Draft Report for Comment

Office of Nuclear Regulatory Research

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Draft Report for Comment

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ABSTRACT

Accident phenomena and offsite consequences of severe reactor accidents have been the subjects of considerable research over the last several decades by the U.S. Nuclear Regulatory Commission (NRC). As a consequence of this research focus, analyses of severe accidents at nuclear power reactors are more detailed, integrated, and realistic than at any time in the past. A desire to leverage this capability to address conservative aspects of previous reactor accident analyses was a major motivating factor in the genesis of the State-of-the-Art Reactor Consequence Analyses (SOARCA) project. By applying modern analysis tools and techniques, the SOARCA project developed a body of knowledge regarding the realistic outcomes of severe nuclear reactor accidents. To accomplish this objective, the SOARCA project's integrated modeling of accident progression and offsite consequences used both state-of-the-art computational analysis tools and best modeling practices drawn from the collective wisdom of the severe accident analysis community. This study has focused on providing a realistic evaluation of accident progression, source term, and offsite consequences for select scenarios for the Peach Bottom Atomic Power Station and Surry Power Station. By using the most current emergency preparedness practices and plant capabilities, as well as the best available modeling, these analyses are more realistic than past analyses. These analyses also consider mitigative measures (e.g., emergency operating procedures, severe accident management guidelines, and Title 10 to the *Code of Federal Regulations* (10 CFR) 50. 54(hh) measures), contributing to a more realistic evaluation.

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ACRONYMS

ac	alternating current
AEC	Atomic Energy Commission
ATWS	anticipated transient without scram
BEF	biological effectiveness factor
BEIR	Biological Effectiveness of Ionizing Radiation
BWR	boiling-water reactor
CDF	core damage frequency
CFR	*Code of Federal Regulations*
CRAC	Calculation of Reactor Accident Consequences
Cs	cesium
CST	condensate storage tank
dc	direct current
DOE	U.S. Department of Energy
ECCS	emergency core cooling system
ECST	emergency condensate storage tank
EOF	emergency operating facility
EOP	emergency operating procedure
EPA	U.S. Environmental Protection Agency
EPR	Evolutionary Power Reactor
EPZ	emergency planning zone
ESBWR	economic simplified boiling-water reactor
ETE	evacuation time estimate
FGR	Federal guidance report
FR	*Federal Register*
GNEP	Global Nuclear Energy Partnership
GNF	Global Nuclear Fuel
gpm	gallons per minute
HTGR	high-temperature gas-cooled reactor
HPCI	high-pressure coolant injection
HPS	Health Physics Society
hr	hour
I	iodine
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
INPO	Institute of Nuclear Power Operations
kg	kilogram
KI	potassium iodide
IPEEE	individual plant examination of external events
ISLOCA	interfacing systems loss-of-coolant accident
LCF	latent cancer fatality
LHSI	low-head safety injection
LNT	linear no-threshold
LOCA	loss-of-coolant accident
LOOP	loss of offsite power

LTSBO	long-term station blackout
LWR	light-water reactor
MACCS2	MELCOR Accident Consequence Code System, Version 2
m/s	meters per second
MOX	mixed oxide
MTU	metric ton of uranium
MW	megawatts
MWd	megawatt days
NCRP	National Council on Radiation Protection and Measurements
NGNP	Next Generation Nuclear Plant
NPP	nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
NRF	National Response Framework
ORNL	Oak Ridge National Laboratory
ORO	offsite response organizations
PGA	peak ground acceleration
PRA	probabilistic risk assessment
PWR	pressurized-water reactor
RCIC	reactor core isolation cooling
RCS	reactor coolant system
RG	regulatory guide
RPV	reactor pressure vessel
RWST	refueling water storage tank
SAMG	severe accident management guideline
SBO	station blackout
SGTR	steam generator tube rupture
SGTS	standby gas treatment system
SNL	Sandia National Laboratories
SOARCA	State-of-the-Art Reactor Consequence Analyses
SPAR	standardized plant analysis risk
SSE	safe shutdown earthquake
SST	siting source term
STCP	Source Term Code Package
STSBO	short-term station blackout
Sv	sieverts
TDAFW	turbine-driven auxiliary feedwater
TID	Technical Information Document
TISGTR	thermally induced steam generator tube rupture
TMI	Three Mile Island
TRANS	transients
TSC	technical support center
UFSAR	updated final safety analysis report
UO_2	uranium dioxide
VHTGR	very high-temperature gas-cooled reactor

EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission (NRC), the nuclear power industry, and the international nuclear energy research community have devoted considerable research over the last several decades to examining severe reactor accident phenomena and offsite consequences. Following the terrorist attacks of 2001, an NRC initiative reassessed severe accident progression and offsite consequences in response to security-related events. These updated analyses incorporated the wealth of accumulated research and used more detailed, integrated, and best-estimate modeling than past analyses. An insight gained from these security assessments was that the NRC needed updated analyses of severe reactor accidents to reflect realistic estimates of the more likely outcomes, considering the current state of plant design and operation and the advances in understanding of severe accident behavior.

The NRC initiated the State-of-the-Art Reactor Consequence Analyses (SOARCA) project to develop best estimates of the offsite radiological health consequences for potential severe reactor accidents for two pilot plants: the Peach Bottom Atomic Power Station in Pennsylvania and the Surry Power Station in Virginia. Peach Bottom is generally representative of U.S. operating reactors using the General Electric boiling-water reactor (BWR) design with a Mark I containment. Surry is generally representative of U.S. operating reactors using the Westinghouse pressurized-water reactor (PWR) design with a large, dry (subatmospheric) containment. SOARCA results, while specific to Peach Bottom and Surry, may be generally applicable to plants with similar designs. Additional work would be needed to confirm this, however, since differences exist in plant-specific designs, procedures, and emergency response characteristics.

The SOARCA project evaluates plant improvements and changes not reflected in earlier NRC publications such as NUREG/CR-2239, “Technical Guidance for Siting Criteria Development,”[1] NUREG-1150, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants,” [2] and WASH-1400, “Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants,” [3]. SOARCA includes system improvements, improvements in training and emergency procedures, offsite emergency response, and security-related improvements, as well as plant changes such as power uprates and higher core burnup. To provide perspective between SOARCA results and more conservative offsite consequence estimates, SOARCA results are compared to NUREG/CR-2239, “Technical Guidance for Siting Criteria Development,” issued in 1982 and referred to in this report as the Siting Study [1]. Specifically, SOARCA results are compared to the Siting Study siting source term 1 (SST1). SST1 assumes severe core damage, loss of all safety systems, and loss of containment after 1.5 hours. The SOARCA report helps the NRC to communicate its current understanding of severe-accident-related aspects of nuclear safety to stakeholders, including Federal, State, and local authorities, licensees, and the general public.

The SOARCA project sought to focus its resources on the more important severe accident scenarios for Peach Bottom and Surry. The project narrowed its approach by using an accident sequence’s possibility of damaging reactor fuel, or core damage frequency (CDF), as a surrogate for risk. The SOARCA scenarios were selected from the results of existing probabilistic risk assessments (PRAs). Core damage sequences from previous staff and licensee PRAs were identified and binned into core damage groups. A core damage group consists of core damage sequences that have similar timing for important severe accident phenomena and similar containment or engineered safety feature operability. It is important to note that each core damage sequence that belongs to a given core damage group is initiated by a specific cause (for example, a seismic event, a fire, or a flood), and that the frequency of each core damage group was estimated by aggregating the CDFs of the individual sequences that belong to the group. This approach was taken to help ensure that the contributions from all core damage sequences were accounted for during the sequence selection process. During the consequence analysis, the core damage groups for station blackouts were analyzed as if they were initiated by a seismic event. This approach was taken because seismically induced equipment failures occur immediately following the seismic event, which produces the severe challenge to the plant. The groups were screened according to their approximate CDFs to identify the most risk significant groups. SOARCA analyzed scenarios with a CDF equal to or greater than 10^{-6} (1 in a million) per reactor-year. SOARCA also sought to analyze scenarios leading to an early failure or bypass of the containment with a CDF equal to or greater than 10^{-7} (1 in 10 million) per reactor-year, since these scenarios have a potential for higher consequences and risk. This approach allowed a more detailed analysis of accident consequences for the more likely, although still remote, accident scenarios.

The staff used updated and benchmarked standardized plant analysis risk (SPAR) models and available plant-specific external events information in the scenario-selection process and identified two major groups of accident scenarios for analysis. The first group common to both Peach Bottom and Surry includes short-term station blackout (STSBO) and long-term station blackout (LTSBO). Both types of SBOs involve a loss of all alternating current (ac) power. The STSBO also involves the loss of turbine-driven systems through loss of direct current (dc) control power or loss of the condensate storage tank and therefore proceeds to core damage more rapidly (hence “short term”). The STSBO has a lower CDF, since it requires a more severe initiating event and more extensive system failures. SBO scenarios can be initiated by external events such as a fire, flood, or earthquake. SOARCA assumes that an SBO is initiated by a seismic event since this is the most extreme case in terms of both the timing and amount of equipment that fails. Notwithstanding the SOARCA scenario screening process, SBO scenarios are commonly identified as important contributors in PRA because of the common cause of failure for both reactor safety systems and containment safety systems.

SOARCA’s second severe accident scenario group, which was identified for Surry only, is the containment bypass scenario. For Surry, two containment bypass scenarios were identified and analyzed. The first bypass scenario is a variant of the STSBO scenario, involving a thermally-induced steam generator tube rupture (TISGTR). The second bypass scenario involves an interfacing systems loss-of-coolant accident (ISLOCA) caused by an unisolated rupture of low-head safety injection piping outside containment. The CDF for the ISLOCA, 3×10^{-8} (3 in 100 million) per reactor-year, falls below the SOARCA screening criterion for bypass events but it is analyzed for completeness because NUREG-1150 identified ISLOCA, in addition to SBO and SGTRs, as principal contributors to mean early and latent cancer fatality risks [2]. This scenario-selection process captured the more important internally and externally initiated core damage scenarios.

SOARCA's analyses were performed with two computer codes, MELCOR for accident progression and the MELCOR Accident Consequence Code System, Version 2 (MACCS2) for offsite consequences. The NRC staff's preparations for the analyses included extensive cooperation from the licensees of Peach Bottom and Surry to develop high-fidelity plant systems models, define operator actions including the most recently developed mitigation actions, and develop models for simulation of site-specific and scenario-specific emergency planning and response. Moreover, in addition to input for model development, licensees provided information on accident scenarios from their PRAs. Through tabletop exercises of the selected scenarios with senior reactor operators, PRA analysts, and other licensee staff, licensees provided input on the timing and nature of the operator actions to mitigate the selected scenarios. The licensee input for each scenario was used to develop assumed timelines of operator actions and equipment configurations for implementing available mitigation measures which include mitigation measures beyond those routinely credited in current PRA models. A human reliability analysis, commonly included in PRAs to represent the reliability of operator actions, was not performed for SOARCA, but instead tabletop exercises, plant walkdowns, simulator runs and other inputs from licensee staff were employed to ensure that operator actions and their timings were correctly modeled.

SOARCA modeled several types of mitigation measures, including those specified in emergency operating procedures (EOPs), severe accident management guidelines (SAMGs), and Title 10 to the *Code of Federal Regulations* (10 CFR) 50.54(hh). The 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 each plant's capability to mitigate events involving a loss of large areas of the plant caused by fire and explosions. To assess the benefits of mitigation measures and to provide a basis for comparison to the past analyses of unmitigated severe accident scenarios, the SOARCA project analyzes the selected scenarios twice: first assuming that the event proceeds unmitigated, and then assuming that mitigation is successful. SOARCA's unmitigated cases assumed neither 10 CFR 50.54(hh) equipment nor a subset of other key operator actions that would prevent core damage were implemented. The subset of operator actions not credited was specific to each individual scenario and included such actions as use of the residual heat removal system for the Surry ISLOCA.

For the LTSBO scenarios for both Peach Bottom and Surry (the most likely severe accident scenario for each plant considered in SOARCA) analyzed assuming no mitigation, core damage begins in 9 to 16 hours, and reactor vessel failure begins at about 20 hours. Offsite radiological release due to containment failure begins at about 20 hours for Peach Bottom (BWR) and at 45 hours for Surry (PWR). The SOARCA analyses therefore show that time may be available for operators to take corrective action and get additional assistance from plant technical support centers even if initial efforts are assumed unsuccessful. For the most rapid events (i.e., the unmitigated STSBO in which core damage may begin in 1 to 3 hours), reactor vessel failure begins at roughly 8 hours, possibly allowing time to restore core cooling and prevent vessel failure. In these cases, containment failure and radiological release begins at about 8 hours for Peach Bottom and at 25 hours for Surry. For the unmitigated Surry ISLOCA, the offsite radiological release begins at about 13 hours and in the other bypass event analyzed, the TISGTR, the radiological release begins at about 3.5 hours but is shown by analyses to be substantially smaller than the 1982 Siting Study SST1 release.

In addition to delayed radiological releases relative to the 1982 Siting Study SST1 case, the SOARCA study demonstrates that the amount of radioactive material released is much smaller as shown in Figures 1 (Iodine-131) and 2 (Cesium-137) below. The Surry ISLOCA iodine release is calculated to be 16 percent of the core inventory, but the results are more generally in the range of 0.5 to 2 percent for iodine and cesium for the other scenarios analyzed. By contrast, the 1982 Siting Study SST1 case calculated an iodine release of 45 percent and a cesium release of 67 percent of the core inventory.

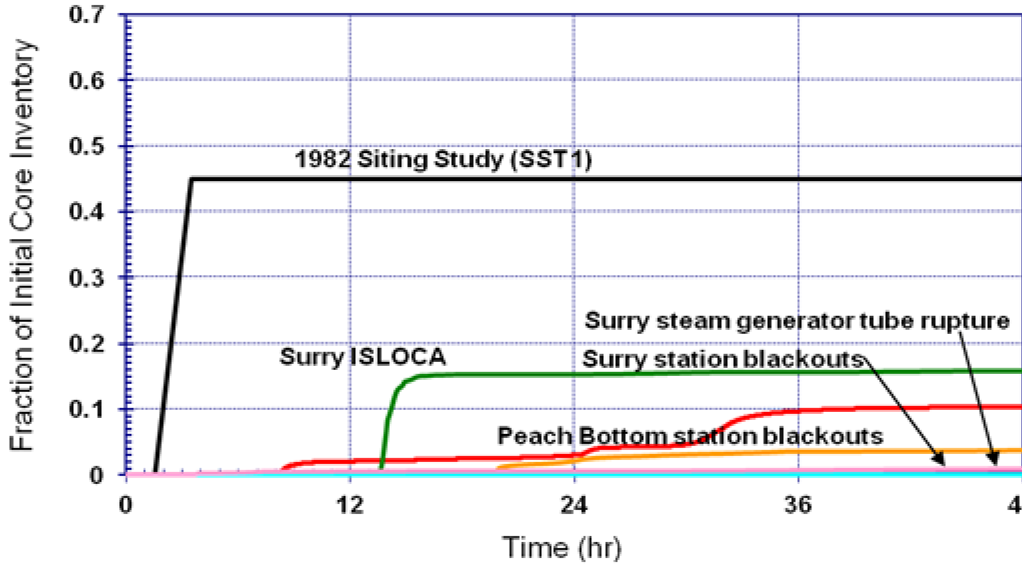

Figure 1 Iodine release to the environment for SOARCA unmitigated scenarios and the 1982 Siting Study SST1 case

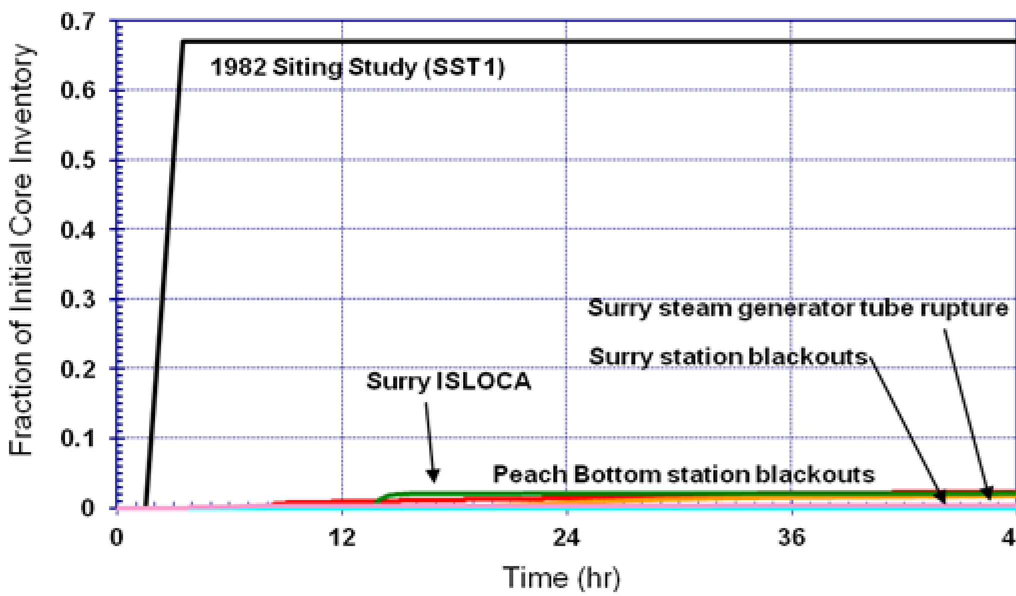

Figure 2 Cesium release to the environment for SOARCA unmitigated scenarios and the 1982 Siting Study SST1 case

Past PRAs and consequence studies showed that sequences involving large early releases were important risk contributors. For example, the PWR SBO with a TISGTR was historically believed to result in a large, relatively early release potentially leading to higher offsite consequences. However, MELCOR analysis of Surry performed for SOARCA shows that the release is small, because other reactor coolant system piping inside containment (i.e., hot leg nozzle) fails soon after the tube rupture and thereby retains the fission products within the containment. Additional work would be needed to determine if this result generally applies for all types of PWRs.

The SOARCA results demonstrate the potential benefits of employing 10 CFR 50.54(hh) mitigation enhancements for the scenarios analyzed. MELCOR analyses were used both to confirm the time available to implement mitigation measures and to confirm that those measures, once taken, are effective in preventing core damage or significantly reducing radiological releases. When successful mitigation is assumed, the MELCOR results indicate no core damage for all scenarios except the Surry STSBO and its TISGTR variant. The security-related mitigation measures that provide alternative ac power and portable diesel-driven pumps are especially helpful in counteracting SBO scenarios. For the Surry STSBO and its TISGTR variant, the mitigation is sufficient to flood the containment through the containment spray system to cover core debris resulting from vessel failure. For the ISLOCA scenario, installed equipment unrelated to 10 CFR 50.54(hh) is effective in preventing core damage owing to the time available for corrective action.

For scenarios that release radioactive material to the environment, MACCS2 uses site-specific weather data to predict the downwind concentration of material in the plume and the resulting population exposures and health effects. The analysis of offsite consequences in SOARCA incorporates the improved modeling capability reflected in the MELCOR and MACCS2 codes as well as detailed site-specific public evacuation models. These models were developed for each scenario based on site-specific emergency preparedness programs and State emergency response plans to reflect timing of onsite and offsite protective action decisions and the evacuation time estimates and road networks at Peach Bottom and Surry. Scenarios that are assumed to be initiated by a seismic event consider the earthquake's impact on implementing emergency plans from loss of infrastructure (i.e., long-span bridges, traffic signals, sirens).

The unmitigated versions of the scenarios analyzed in SOARCA have lower risk of early fatalities than calculated in the 1982 Siting Study SST1 case. SOARCA's analyses show essentially zero risk of early fatalities. Early fatality risk was calculated to be $\sim 10^{-14}$ for the unmitigated Surry ISLOCA (for the area within 1 mile of Surry's exclusion area boundary) and zero for all other SOARCA scenarios. In comparison, 92 early fatalities for Peach Bottom and 45 early fatalities for Surry were calculated for the SST1 case in the 1982 Siting Study.

SOARCA results indicate that bypass events (e.g., Surry ISLOCA) do not pose a higher scenario-specific latent cancer fatality risk than non-bypass events (e.g., Surry SBO). While consequences are greater when the bypass scenario happens, this is offset by the scenario being less likely to happen. SOARCA reinforces the importance of external events relative to internal events and the need to continue ongoing work related to external events risk assessment.

Offsite radiological consequences were calculated for each scenario expressed as the average individual likelihood of an early fatality and latent cancer fatality. Tables 1 (Peach Bottom) and 2 (Surry) show, for both mitigated and unmitigated cases, conditional (on the occurrence of the core damage scenario) scenario-specific probabilities of a latent cancer fatality for an individual located within 10 miles of the plant. Tables 1 and 2 show the results using the linear no-threshold (LNT) dose-response model, which assumes that the health risk is directly proportional to the exposure and even the smallest radiation exposure carries some risk. The tables also provide the scenario-specific latent cancer fatality risk for an individual located within 10 miles of the plant, taking into account the scenario's core damage frequency.

Table 1 Offsite Consequence Results for Peach Bottom Scenarios Assuming Linear No-Threshold (LNT) Dose-Response Model

Scenario	Core damage frequency (CDF) (per reactor-year)	Mitigated		Unmitigated	
		Conditional scenario-specific probability of latent cancer fatality for an individual located within 10 miles	Scenario-specific risk (CDF x Conditional) of latent cancer fatality for an individual located within 10 miles (per reactor-year)	Conditional scenario-specific probability of latent cancer fatality for an individual located within 10 miles	Scenario-specific risk (CDF x Conditional) of latent cancer fatality for an individual located within 10 miles (per reactor-year)
Long-term SBO	3×10^{-6}	No Core Damage		9×10^{-5}	~ 3×10^{-10} ***
Short-term SBO	3×10^{-7}	No Core Damage **		2×10^{-4}	~ 6×10^{-11} ***
Short-term SBO with RCIC Blackstart*	3×10^{-7}			7×10^{-5}	~ 2×10^{-11} ***

* Blackstart of the reactor core isolation cooling (RCIC) system refers to starting RCIC without any ac or dc control power. Blackrun of RCIC refers to the long-term operation of RCIC without electricity, once it has been started. This typically involves using a portable generator to supply power to indications such as reactor pressure vessel (RPV) level to allow the operator to manually adjust RCIC flow to prevent RPV overfill and flooding of the RCIC turbine.

** If the RCIC system is successfully controlled (i.e., successful blackstart and blackrun) then both mitigated Short-term SBO scenarios would be functionally similar to the mitigated Long-term SBO (i.e., no core damage). This was qualitatively determined based on the timing and equipment availabilities from the other SBO analyses.

*** Estimated risks below 1×10^{-7} per reactor year should be viewed with caution because of the potential impact of events not studied in the analyses and the inherent uncertainty in very small calculated numbers.

Table 2 Offsite Consequence Results for Surry Scenarios Assuming LNT Dose-Response Model

Scenario	Core damage frequency [CDF] (per reactor-year)	Mitigated		Unmitigated	
		Conditional scenario-specific probability of latent cancer fatality for an individual located within 10 miles	Scenario-specific risk [CDF x Conditional] of latent cancer fatality for an individual located within 10 miles (per reactor-year)	Conditional scenario-specific probability of latent cancer fatality for an individual located within 10 miles	Scenario-specific risk [CDF x Conditional] of latent cancer fatality for an individual located within 10 miles (per reactor-year)
Long-term SBO	2×10^{-5}	No Core Damage		5×10^{-5}	~ 7×10^{-10} ***
Short-term SBO	2×10^{-6}	No Containment Failure *		9×10^{-5}	~ 1×10^{-10} ***
Short-term SBO with TISGTR	4×10^{-7}	3×10^{-4} **	~ 1×10^{-10} ***	3×10^{-4}	~ 1×10^{-10} ***
Interfacing systems LOCA	3×10^{-8}	No Core Damage		3×10^{-4}	~ 9×10^{-12} ***

* Accident progression calculations showed that source terms in the mitigated case are smaller than in the unmitigated case. Offsite consequence calculations were not run, since the containment fails at about 66 hours. A review of available resources and emergency plans shows that adequate mitigation measures could be brought onsite within 24 hours and connected and functioning within 48 hours. Therefore 66 hours would allow ample time for mitigation through measures transported from offsite.

** Containment failure is delayed by about 46 hours in the mitigated case relative to the unmitigated case. Rounding to one significant figure shows conditional LCF probabilities of 3×10^{-4} for both mitigated and unmitigated cases, however the original values were 2.8×10^{-4} for the mitigated case and 3.2×10^{-4} for the unmitigated case.

*** Estimated risks below 1×10^{-7} per reactor year should be viewed with caution because of the potential impact of events not studied in the analyses and the inherent uncertainty in very small calculated numbers.

LCF risks using alternate dose-response models, as well as LCF risks for circular areas out to a radius of 50 miles, are also presented. Using a dose-response model that truncates annual doses below normal background levels (including medical exposures) results in a further reduction to the latent cancer fatality risks (by a factor of 100 for smaller releases and a factor of 3 for larger releases). Latent cancer fatality risk calculations are generally dominated by long-term exposure to small annual doses (~500 mrem per year corresponding to state return criteria) by evacuees returning to their homes after the accident and being exposed to residual radiation over a long period of time. SOARCA's calculated LCF risk results are smaller than extrapolations of 1982 Siting Study SST1 LCF risk results. However, the difference diminishes when considering larger areas, out to a distance of 50 miles from the plant.

Figure 3 compares SOARCA's scenario-specific latent cancer fatality risks for an individual within 10 miles of the plant to the NRC Safety Goal [72] and to an extrapolation of the 1982 Siting Study SST1[1] results.

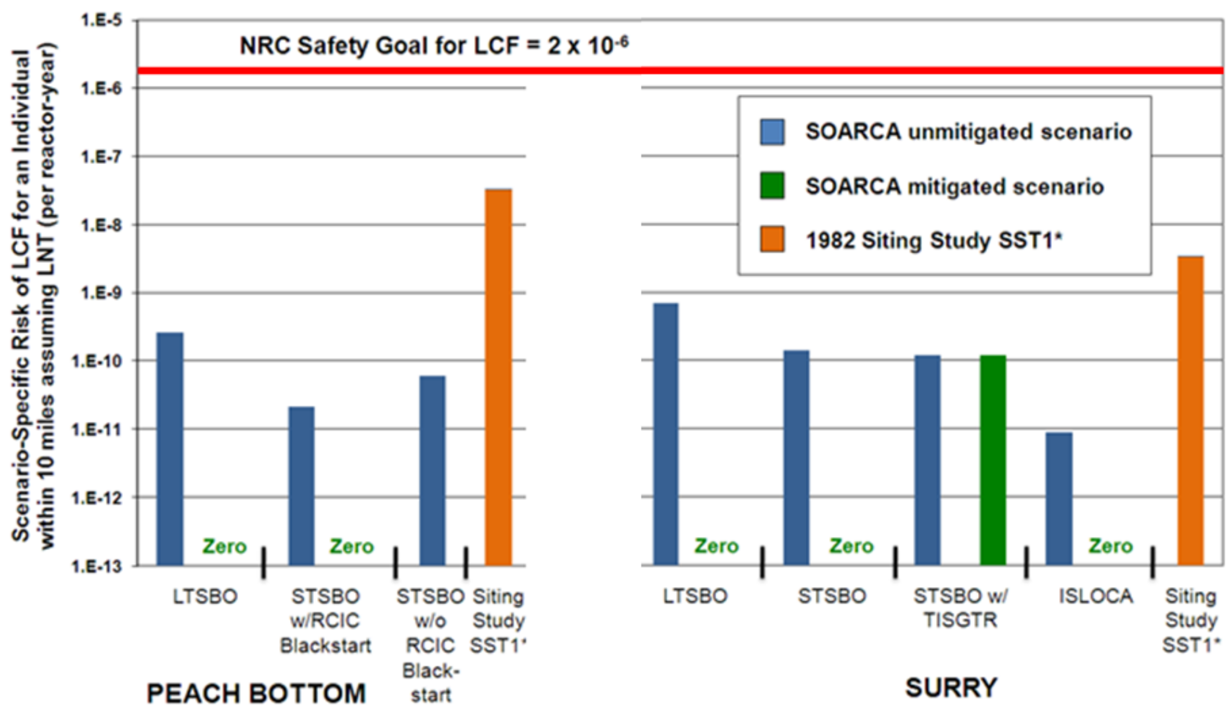

Figure 3 Comparison of individual LCF risk results for SOARCA mitigated and unmitigated scenarios to the NRC Safety Goal and to extrapolations of the 1982 Siting Study SST1 (plotted on logarithmic scale)

The NRC Safety Goal for latent cancer fatality risk from nuclear power plant operation (i.e., 2×10^{-6} or two in one million) is set 1,000 times lower than the sum of cancer fatality risks

1 The Siting Study did not calculate LCF risks. Therefore, to compare the Siting Study SST1 case to LCF results for SOARCA, the SST1 source term was put into the MACCS2 offsite consequence code files for the Peach Bottom and Surry unmitigated STSBO calculations.

resulting from all other causes (i.e., $2x10^{-3}$ or two in one thousand). The calculated cancer fatality risks from the selected, important scenarios analyzed in SOARCA are thousands of times lower than the NRC Safety Goal and millions of times lower than the general U.S. cancer fatality risk [73].

Comparisons of SOARCA's calculated LCF risks to the NRC Safety Goal [72] and the average annual US cancer fatality risk from all causes [73] are provided to give context that may help the reader to understand the contribution to cancer risks from these nuclear power plant accident scenarios. However, such comparisons have limitations for which the reader should be aware. Relative to the safety goal comparison, the safety goal is intended to encompass all accident scenarios. SOARCA does not examine all scenarios typically considered in a PRA, even though it includes the important scenarios. In fact, any analytical technique, including PRAs, will have inherent limitations of scope and method. As a result, comparison of SOARCA's scenario-specific calculated LCF risks to the NRC Safety Goal is necessarily incomplete. However, it is intended to show that adding multiple scenarios' low risk results in the ~ 10^{-10} range to approximate a summary risk from all scenarios, would yield a summary result that is also below the NRC Safety Goal of $2x10^{-6}$ or two in one million.

Relative to the U.S. average individual risk of a cancer fatality comparison, the sources of an individual's cancer risk include a complex combination of age, genetics, lifestyle choices, and other environmental factors whereas the consequences from a severe accident at a nuclear plant are involuntary and unlikely to be experienced by most individuals.

The SOARCA analyses show that emergency response programs, implemented as planned and practiced, reduce the scenario-specific risk of health consequences among the public during a severe reactor accident. Sensitivity analyses of seismic impacts on site-specific emergency response (e.g., loss of bridges, traffic signals, and delayed notification) at Peach Bottom and Surry do not significantly affect LCF risk.

In summary, the staff believes SOARCA has achieved its objective to develop a body of knowledge regarding detailed, integrated, state-of-the-art modeling of the most important severe accident scenarios for Peach Bottom and Surry. 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 the 1982 Siting Study SST1 case. As a result, the calculated risks of public health consequences from severe accidents modeled in SOARCA are very small.

The SOARCA study was nearing completion when the Fukushima Daiichi accident occurred on March 11, 2011. The Fukushima accident has many similarities and differences with some of the Peach Bottom severe accident scenarios analyzed in SOARCA. While there are significant gaps in information and uncertainties regarding what occurred in the Fukushima reactors, an appendix to this report compares and contrasts the SOARCA study and the Fukushima accident based on currently available information for the following topics: (1) operation of the RCIC system, (2) hydrogen release and combustion, (3) 48-hour truncation of releases in SOARCA, (4) multiunit risk, and (5) spent fuel pool risk.

1.0 INTRODUCTION

This document describes the U.S. Nuclear Regulatory Commission's (NRC's) state-of-the-art, realistic assessment of the accident progression, radiological releases, and offsite consequences for important severe accident sequences.

The overall objective of the State-of-the-Art Reactor Consequence Analyses (SOARCA) project is to develop a body of knowledge on the realistic outcomes of severe reactor accidents. The results from the SOARCA project to date provide an updated reference of the likely outcomes of severe reactor accidents at the Peach Bottom and Surry nuclear power sites, based on the most current emergency preparedness and plant capabilities. The NRC also anticipates that the study will be a resource for future modeling improvements and verification efforts.

1.1 Background

The evaluation of accident phenomena and offsite consequences of severe reactor accidents has been the subject of considerable research. Most recently, with Commission guidance and as part of plant security assessments, updated analyses of severe accident progression and offsite consequences were completed using the wealth of accumulated research. These analyses are more detailed (in terms of the fidelity of the representation and resolution of facilities and emergency response), realistic (in terms of the use of currently accepted phenomenological models and procedures), and integrated (in terms of the intimate coupling between accident progression and offsite consequence models).

The results of those security-related studies confirmed and quantified what was suspected but not well-quantified—namely, that some past studies were conservative to the point that predictions were not useful for characterizing results. The communication of risk attributable to severe reactor accidents should properly consider realistic estimates of the more likely outcomes and should reflect both the many improvements and changes to plants and the advances in understanding of severe accident behavior.

In addition to the improvements in understanding and calculational capabilities that have resulted from these studies, numerous influential changes have occurred in the training of operating personnel and the increased use of plant-specific capabilities. These changes include the following:

- The transition from event-based to symptom-based emergency operating procedures (EOPs) for the boiling-water reactor (BWR) and pressurized-water reactor (PWR) designs.
- The performance and maintenance of plant-specific probabilistic risk assessments (PRAs) that cover the spectrum of accident scenarios.
- The implementation of plant-specific, full-scope control room simulators to train operators.

- An industrywide technical basis, owners-group-specific guidance, and plant-specific implementation of the severe accident management guidelines (SAMGs).

- Additional safety enhancements, described in Title 10, Section 50.54(hh) of the *Code of Federal Regulations* (10 CFR 50.54(hh)). These enhancements are intended to be used to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities under the circumstances associated with loss of large areas of the plant due to explosions or fire, to include strategies in the following areas: (i) fire fighting; (ii) operations to mitigate fuel damage; and (iii) actions to minimize radiological release. For the SOARCA scenarios, successful implementation of this equipment and procedures would prevent core damage or delay or prevent the release.

- Improved phenomenological understanding of influential processes such as the following:

  – in-vessel steam explosions
  – Mark I containment drywell shell attack
  – dominant chemical forms for fission products
  – direct containment heating
  – hot-leg creep rupture
  – reactor pressure vessel (RPV) failure and molten core-concrete interactions

Additional changes in plant operation have occurred over time, including the following:

- power uprates
- higher core burnups

1.2 Objective

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 [1].

1.3 Approach

The approach was to use the detailed, integrated, phenomenological modeling of accident progression (reactor and containment thermal-hydraulic and radionuclide response) that is embodied in the MELCOR code, coupled with modeling of offsite consequences with the MELCOR Accident Consequence Code System, Version 2 (MACCS2) code, to predict the likely outcomes for the more significant, albeit still remote, core melt accidents. The basis for the selection of the events for analysis included insights from past and current PRAs and from research on accident behavior and important failure modes. The selection of events for quantification also properly included probability, to focus on more likely and important contributors.

Figure 1 illustrates the four main elements of SOARCA (i.e., scenario selection, mitigative measures analysis, accident progression and source term, and offsite radiological consequences).

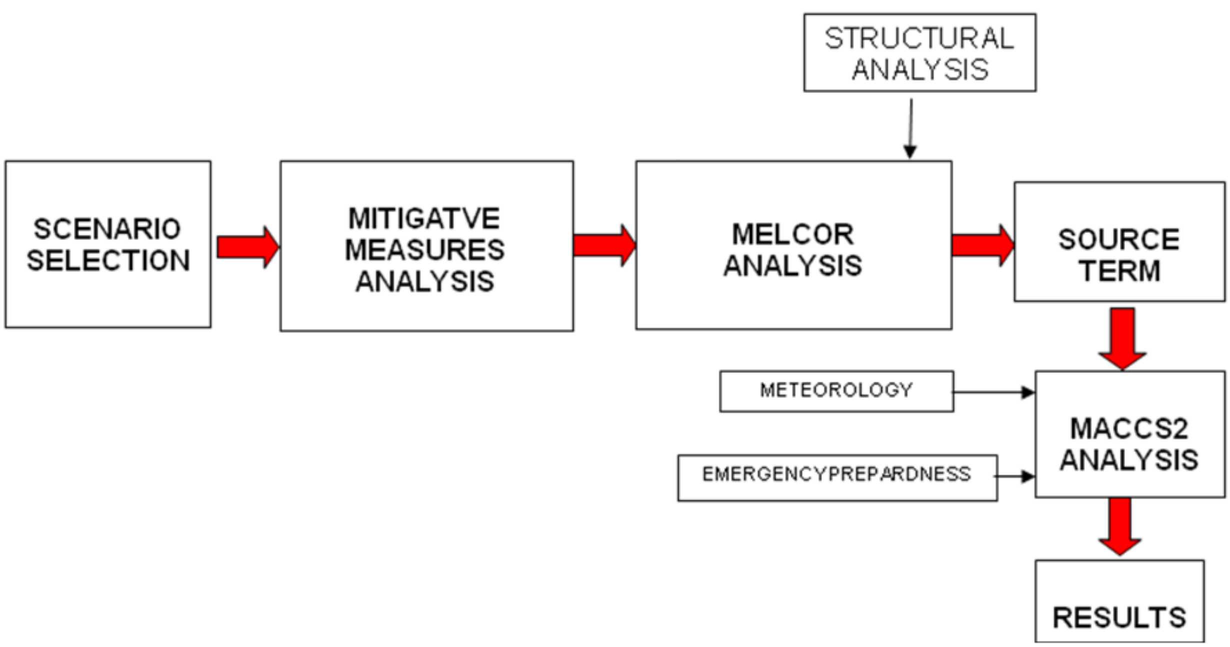

Figure 1 The State-of-the-Art Reactor Consequence Analyses process

SOARCA provides a new and useful tool to, at this juncture, focus on specific important events and quantify the plant and offsite response rigorously and realistically. This approach can complement and supplement other analytical methods to efficiently and explicitly address the benefits of additional mitigation in further reducing the likelihood of core damage and offsite consequences. The offsite consequence analyses were performed on a site-specific basis (reflecting site-specific population distributions, weather, and emergency preparedness). Selection of events considered individual plant examinations,[2] individual plant examinations of external events (IPEEEs), standardized plant analysis risk (SPAR) models, and NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," issued 1990 [2]. The plant modeling included information related to system and procedural plant improvements that were incorporated as part of the industry's response to the NRC's security initiatives (e.g., the purchase and development of procedures for diesel-driven pumps in response to 10 CFR 50.54(hh) requirements), as well as necessary plant information.

1.4 Historical Perspectives

The following sections describe some of the important historical studies that preceded the SOARCA project.

1.4.1 WASH-1400 (NUREG-75/014), "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," 1975

In the summer of 1972, the Atomic Energy Commission (AEC) initiated a major probabilistic study, "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants" [3]. Professor Norman C. Rasmussen of the Massachusetts Institute of Technology served as the study director. Saul Levine of the AEC served as staff director of the AEC employees who performed the study with the aid of many contractors and consultants.

The study team attempted to estimate the potential effects of light-water reactor (LWR) accidents on public health and safety. The report analyzed in detail one BWR, Peach Bottom Unit 2, and one PWR, Surry Unit 1, to estimate the likelihood and consequences of potential accidents. The team chose these plants, because they were the largest plants of each type that were about to start operation.

The study's purpose was to quantify the risks to the general public from commercial nuclear power plant (NPP) operation and to compare those risks with nonnuclear risks to provide perspective. This required identification, quantification, and phenomenological analysis of a wide range of low-frequency, relatively high-consequence scenarios that had not previously been considered in much detail. The introduction at this point of the concept of "scenario" is significant; as noted above, many design assessments simply look at system reliability (success probability), given a design-basis challenge. The review of nuclear plant license applications did essentially this, culminating in findings that specific complements of safety systems were single-failure proof for selected design-basis events. Going well beyond this, WASH-1400

[2] As requested by the NRC in Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities," dated November 23, 1988, the utilities conducted risk analyses that considered the unique aspects of a particular NPP, identifying the specific vulnerabilities of the plant to severe accidents.

modeled scenarios leading to large radiological releases from each of the commercial NPPs considered. It considered highly complex scenarios involving the success and failure of many and diverse systems within a given scenario, as well as operator actions and phenomenological events.

The team adapted methods previously used by the U.S. Department of Defense and the National Aeronautics and Space Administration to predict the effect of failures of small components in large, complex systems. The overall methodology, PRA, is still used today.

The team first identified events that could potentially lead to core damage. It then used event trees to delineate possible sequences of successes or failures of systems provided to prevent core meltdown or the release of radionuclides, or both. Using fault trees, the team estimated the probabilities of system failures from available data on the reliability of system components. With these techniques, thousands of possible core melt accident sequences were assessed for their occurrence probabilities. Computational models developed as part of the overall effort calculated the public health and economic consequences of the identified severe accidents.

The insights gained from WASH-1400 included (1) "the possible consequences of potential reactor accidents are predicted to be no larger, and in many cases much smaller, than those of nonnuclear accidents," (2) "the likelihood of reactor accidents is much smaller than that of many non-nuclear accidents having similar consequences. All non-nuclear accidents examined in this study, including fires, explosions, toxic chemical releases, dam failures, airplane crashes, earthquakes, hurricanes and tornadoes, are much more likely to occur and can have consequences comparable to, or larger than, those of nuclear accidents," and (3) "non-nuclear events are about 10,000 times more likely to produce large numbers of fatalities than nuclear plants."

While the risks from nuclear power appear to be very low, the Reactor Safety Study (WASH-1400) did indicate that core melt accidents were more likely than previously thought (approximately 5×10^{-5} per reactor-year for Surry and Peach Bottom[3]), and that LWR risks are mainly attributable to core melt accidents. The Reactor Safety Study also demonstrated the wide variety of accident sequences (initiators and ensuing equipment failures or operator errors or both) that can cause core melt. In particular, the report indicated that, for the plants analyzed, accidents initiated by transients or small loss-of-coolant accidents (LOCAs) were more likely to cause core melt than the traditional large design-basis LOCAs.

In addition to providing some quantitative perspective on severe accident risks, other significant WASH-1400 results helped increase the application of PRAs in the commercial nuclear power arena. They showed, for example, that some of the more frequent, less severe initiating events (e.g., "transients") lead to severe accidents at higher expected frequencies than do some of the less frequent, more severe initiating events (e.g., very large pipe breaks). This led to the beginning of the understanding of the level of design detail that a PRA must include, if the scenario set is to support useful findings (e.g., consideration of support systems and

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3 This value is derived from the following statement in the WASH-1400 Executive Summary: "The [probability of melting the core] value obtained was about one 1 in 20,000 per reactor per year."

environmental conditions). Following the severe core damage event at Three Mile Island in 1979, application of these insights gained momentum within the nuclear safety community, leading eventually to a PRA-informed reexamination of the allocation of licensee and regulatory safety resources. In the 1980s, this process led to some significant adjustments to safety priorities at NPPs; since the 1990s, the NRC has refocused its regulations on areas of plant safety where that attention is more risk important.

1.4.2 NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," 1982

The NRC contracted with Sandia National Laboratories (SNL) to develop a technical guidance report for siting future reactors [1]. The agency requested guidance on (1) criteria for population density and distribution surrounding future sites and (2) standoff distances of plants from offsite hazards.

Because the work was primarily focused toward the development of generic siting criteria, uncoupled from specific plant design, five types of accidents, with assumed representative radiological source terms, were imposed on each plant in the 91-site study. The accidents or "siting source term events" (SST events) were to be derived from the previous Reactor Safety Study (WASH-1400) [3], and each SST event would be assumed identical regardless of plant design.

(1) SST1—Severe core damage. All safety systems and containment are lost after 1.5 hours.

(2) SST2—Severe core damage. Containment systems (e.g., sprays, suppression pools) function to reduce radioactive release, but containment leakage is large after 3 hours.

(3) SST3—Severe core damage. Containment systems function, but there is small containment leakage (1 percent per day) after 1 hour.

(4) SST4—Modest core damage. Containment systems function but there is small containment leakage after ½ hour.

(5) SST5—Limited core damage. Containment functions as designed with minimal leakage.

The early fatality results for most of the 91 sites were similar because of the low population density close to the sites. Using the extremely large and rapid SST1 radiological source term with a population density of 50 persons per square mile resulted in 47 to 140 early fatalities and 730 to 860 latent cancer fatalities (LCFs). For the release represented by SST2 events, the mean values from typical plants were zero early fatalities and 95 to 140 LCFs.

1.4.3 NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," 1990

NUREG-1150 [2] documents the results of an extensive NRC-sponsored PRA. The study examined five plants representative of classes of reactor and containment designs to give an understanding of risks for these particular plants. Selected insights regarding the classes of plants were also obtained in the study. The improved PRA methodology used in the

NUREG-1150 study greatly enhanced the understanding of risk at NPPs and is considered a significantly updated and improved revision to the Reactor Safety Study [3]. One improvement was the specific inclusion of an uncertainty estimate for the core damage frequency (CDF) and source term portions of the study. This uncertainty estimate was based on extensive use of expert elicitation. For the offsite consequence portion of the study, random weather sampling addressed the uncertainty in health effects caused by weather variability.

The following five NPPs were analyzed in NUREG-1150:

(1) Unit 1 of the Surry Power Station, a Westinghouse-designed, three-loop PWR reactor in a large, dry, subatmospheric containment building located near Williamsburg, VA

(2) Unit 1 of the Zion Nuclear Power Plant, a Westinghouse-designed, four-loop PWR reactor in a large, dry containment building located near Chicago, IL

(3) Unit 1 of the Sequoyah Nuclear Power Plant, a Westinghouse-designed, four-loop PWR reactor in an ice condenser containment building located near Chattanooga, TN

(4) Unit 2 of the Peach Bottom Atomic Power Station, a General Electric-designed, BWR-4 reactor in a Mark I containment building located near Lancaster, PA

(5) Unit 1 of the Grand Gulf Nuclear Station, a General Electric-designed, BWR-6 reactor in a Mark III containment building located near Vicksburg, MS

The various accident sequences that contribute to the CDF from internal initiators can be grouped by common factors into categories. NUREG-1150 uses the accident categories depicted in Table 1 below: station blackout (SBO), anticipated transients without scram (ATWS), other transients (TRANS), interfacing system LOCAs (SG/IF Sys), and other LOCAs. The selection of such categories is not unique but merely a convenient way to group the results.

Table 1 Summary of Core Damage Frequency from NUREG-1150

	Internal Initiators						**External Initiators**
Plant Name	SBO	ATWS	TRANS	SG/IF Sys	LOCA†	Core Damage Total/yr	Fire & Seismic
Surry	2.7×10^{-5}	1.6×10^{-6}	2.0×10^{-6}	3.4×10^{-6}	6.0×10^{-6}	4.0×10^{-5}	2.6×10^{-5}
Peach Bottom	2.2×10^{-6}	1.9×10^{-6}	1.4×10^{-7}	-	2.6×10^{-7}	4.5×10^{-6}	2.3×10^{-5}

† The LOCA category shown here includes LOCAs that are initiated by pipe break events. Transient-induced LOCAs are included under the other categories.

1.5 Scope

The central focus of the SOARCA project was to introduce the use of a detailed, best estimate, self-consistent quantification of scenarios based on current scientific knowledge and plant capabilities. The essence of the analysis methodology is the application of the integrated severe accident progression modeling tool, the MELCOR code. The analysis used an improved offsite consequence (MACCS2) code, including both improved code input and updated scenario-specific emergency response. Because the priority of this work was to bring more detailed, best estimate, and consistent analytical modeling to bear in determining realistic outcomes of severe accident scenarios, the benefits of this state-of-the-art modeling could most efficiently be demonstrated by applying these methods to a set of the more important severe accident scenarios. Thus, the project elected to limit its analysis to a set of important accident scenarios considering both likelihood and potential consequences. The scenarios that were eventually selected (e.g., SBO, interfacing systems loss-of-coolant accident (ISLOCA), thermally induced steam generator tube rupture (TISGTR)) are, in fact, scenarios that were also considered to be important in recent and past probabilistic assessments.

The following several classes of accident events were not considered as part of the SOARCA project:

- multiunit accidents
- low-power and shutdown accidents
- extreme seismic events that lead directly to gross containment failure with simultaneous reactor core damage
- spent fuel pool accidents
- security events

Multiunit accidents (events leading to reactor core damage at multiple units on the same site) could be caused by certain initiators such as an earthquake. Most PRAs developed to date do not explicitly consider multiunit accidents, because the NRC policy is to apply the Commission's "Safety Goals for the Operation of Nuclear Power Plants" (51 *Federal Register* (FR) 28044) [4] and subsidiary risk acceptance guidelines (see Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" [5]) on a "per reactor" basis. Therefore, multiunit accidents were not evaluated in the SOARCA project. The results of the unmitigated scenario analyses in SOARCA suggest that consideration of multiunit events would not substantially alter the study findings regarding low individual risk, but explicit analysis would be required to confirm the conclusion.

Low-power and shutdown accidents are potentially significant, because the plant configuration is altered—the containment may be open and the reactor safety systems may be realigned. However, offsetting mitigating attributes include a potentially much smaller decay heat level and low pressure that allows for easier cooling of the reactor fuel. In this area, SOARCA has focused on the accidents that historically have received the most attention—the accidents initiated at full power. Also, one of the objectives was to provide an updated quantification of risk from past studies such as the Siting Study [1], and that study similarly was confined to full-power reactor events.

The SOARCA study excluded extreme seismic events that involve failure of the containment and lead to 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.

Spent fuel pool accidents can contribute to overall risk associated with nuclear reactors, because significant quantities of spent fuel are stored onsite in such pools. Past NRC studies, including NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," issued February 2001 [62], would suggest that risk from the most severe spent fuel pool accidents is low, yet the consequences of the release of a large inventory of cesium (Cs) and other radioisotopes could be serious. Since that time, the NRC has undertaken substantial analytical and experimental research to improve the modeling of spent fuel pool accidents, as well as research to identify significant improvements to spent fuel pool safety, as part of the NRC's security-related research following the terrorist attacks of September 11, 2001. Based on the results of this research, the NRC concludes that spent fuel pool risk, which was assessed very conservatively in past studies such as NUREG-1738, is now much lower, based on both the new physical safety improvements required by the NRC and the improved modeling capability. Therefore, when developing the SOARCA project, the NRC elected to exclude spent fuel pool accidents from its scope.

The NRC did not include security events as part of SOARCA to avoid providing any specific information that may materially assist in planning or carrying out a terrorist attack on an NPP. However, the NRC has stated that the security-related studies conducted after September 11, 2001, led it to conclude that previous risk studies used conservative radionuclide source terms and that plant improvements plus improved modeling would confirm that radionuclide releases and early fatalities were substantially smaller than suggested by earlier studies.

Offsite consequences of severe nuclear reactor accidents could include economic and environmental damage in addition to harmful effects on human health. SOARCA calculates offsite consequences in terms of the risks of human fatalities for the specific scenarios. These risks are quantified as the individual risk of an early fatality and the individual risk of a latent cancer fatality. This enables comparison of SOARCA's results to the NRC Safety Goal and to the 1982 Siting Study's results.

Offsite consequences of severe reactor accidents could include economic and environmental consequences, the temporary or permanent displacement of local populations and businesses, and harmful effects on human health. SOARCA calculates offsite consequences in terms of the risks of human fatalities for the specific scenarios. These risks are quantified as the individual risk of an early fatality and the individual risk of a latent cancer fatality. This enables comparison of SOARCA's results to the NRC Safety Goal [72] and to the 1982 Siting Study's results [1].

1.6 Basis of Accident Selection

In the selection of important sequences, the SOARCA project ideally would have included those sequences found to be important to risk as demonstrated by a full-scope Level 3 PRA, which is in assessment of risk of offsite consequences in the event of a severe accident causing release of radioactive material to the environment. In practice, that was not feasible, because no current full-scope Level 3 PRAs (considering both internal and external events) were generally available to draw upon. However, the preponderance of Level 1 PRA information, combined with insights on severe accident behavior, is available on dominant core damage sequences, especially internal event sequences. This information, combined with the NRC's understanding of containment loadings and failure mechanisms, and together with radionuclide release, transport, and deposition, allows the use of CDF as a surrogate criterion for risk. Thus, for SOARCA, the project team elected to analyze sequences with a CDF greater than 10^{-6} per reactor-year. In addition, the SOARCA team included sequences that have an inherent potential for higher consequences (and risk) with a lower CDF (i.e., those with a frequency greater than 10^{-7} per reactor-year). Such sequences would be associated with events involving containment bypass or leading to an early failure of the containment. By adopting these criteria, the SOARCA team is reasonably assured that the more probable and important core melt sequences will be captured. Further, SOARCA includes certain scenarios that had CDFs lower than the screening criteria, because of their historical significance. Thus, the selection of scenarios has a more generic application to plants with designs similar to Peach Bottom and Surry.

1.7 Mitigated and Unmitigated Cases

An important objective of the SOARCA project was to assess the impact of severe accident mitigative features and reactor operator actions in mitigating an accident. This was done by evaluating in detail the operator actions and equipment that may be available (including 10 CFR 50.54(hh) equipment).

Early in the project (2007), SOARCA staff visited the Peach Bottom Atomic Power Station and the Surry Power Station. During the visits, tabletop exercises were conducted for each scenario. Participants included plant senior reactor operators and PRA analysts. SOARCA staff provided initial and boundary conditions, elicited how plant staff would respond, and, through the tabletop exercises, developed a timeline of operator actions for each scenario. These assessments of mitigative measures were qualitative but, nonetheless, consisted of detailed scenario-specific consideration of systems and operations, based on licensee-identified mitigative measures from EOPs, SAMGs, 10 CFR 50.54(hh) measures, assistance from the technical support center (TSC), and other severe accident guidelines that are applicable to and determined to be available during a specific scenario. The assessment of mitigation systems provided the basis for the assumptions on availability, capability, and timing used as input into the MELCOR analyses. For scenarios involving a seismic initiator, operator response times were lengthened to reflect the severity of the seismic event.

A traditional human reliability assessment has not been performed to quantify the probabilities of plant personnel succeeding in implementing these measures. However, the NRC issued 10 CFR 50.54(hh) requiring plant licensees to possess the equipment, develop the strategies, and train plant personnel to implement these mitigative measures. The 10 CFR 50.54(hh) measures are the result of a major effort by industry and the NRC in the 2004–2008 timeframe to develop means to mitigate events involving a loss of large areas of the plant caused by fire and explosions. These mitigation measures were implemented by each plant on a per site basis rather than a per reactor basis, however some licensees have indicated plans to purchase additional equipment for the other unit. These measures are new and diverse and include the following major elements:

- procedures for manually operating turbine-driven injection (reactor core isolation cooling (RCIC) and turbine-driven auxiliary feedwater (TDAFW)) systems
- portable diesel-driven pumps for injecting into the reactor coolant system (RCS) (BWR) and steam generators (PWR)
- alternative means to depressurize
- portable power supplies for critical instrumentation

The assessment of mitigation measures has continued to receive attention since the initial assessment conducted with plant staff. The SOARCA team conducted additional site visits and system walkdowns in 2007, 2010, and 2011, with licensee personnel specifically reviewing the mitigation steps. The team used the results of accident progression calculations to characterize anticipated changes in plant conditions and describe the signatures of measurable parameters. It then estimated the time needed to assemble necessary personnel, tools, and equipment; align and start components; and establish a desired operating condition. SOARCA staff conducted followup site visits in June and August 2010 to explicitly address RCIC blackstart and blackrun for short-term station blackout (STSBO) and manual operation of TDAFW. The site visits included a review of RCIC blackstart and blackrun procedures, additional tabletop exercises to refine the PWR STSBO timeline, plant walkdowns of equipment areas, and detailed reviews of procedures. For the ISLOCA scenario, the licensee also had reactor operators use EOPs in a plant simulator to ensure timing for operator actions to be used in the SOARCA MELCOR calculations was accurate and reasonable.

For each scenario and the mitigation measures identified, the team conducted detailed accident progression analyses to assess the efficacy of those measures. For each scenario, it also performed accident progression and offsite consequence analyses, assuming key mitigative measures were not taken, to demonstrate the relative importance and significance of those measures and to allow comparison of offsite consequence predictions with earlier studies.

For each scenario, the project identified applicable mitigative measures that are potentially available (not eliminated by initial conditions). The systems and operations analyses were based on the initial conditions and anticipated subsequent failures to do the following:

- verify the availability of the primary system
- determine the availability of support systems and equipment
- determine time estimates for implementation

Based on these scenario specifications, the team used MELCOR to determine the effectiveness of those mitigative measures that are expected to be available at a given time.

1.8 Uncertainty Analysis

The SOARCA project included a number of sensitivity studies to examine issues associated with accident progression, mitigation, and offsite consequences for the accident scenarios of interest. The objective of these sensitivity studies was to examine specific issues and ensure the robustness of the conclusions documented in this report. Single sensitivity studies, however, do not form a complete picture of the uncertainty associated with accident progression and offsite consequence modeling. Such a picture requires a more comprehensive and integrated evaluation of modeling uncertainties.

A follow-on uncertainty study will evaluate the impact of uncertainty by randomly sampling distributions for key model parameters that were considered to have a potential impact on the offsite consequences. The intended purpose of this uncertainty study is to develop insight into the overall uncertainty of the SOARCA results on scenario-specific risk to the combined and integrated uncertainty in accident progression (MELCOR) and offsite health effects (MACCS2) modeling. By addressing key MELCOR and MACCS2 modeling uncertainties in an integrated fashion, the SOARCA team believes it will further its understanding of the importance of this modeling on risk and thereby reveal where improvements in understanding are likely to be of benefit. (It will not address uncertainty in the scenario frequency.) Of principal interest is a comparison of the mean value, as determined by the uncertainty analysis, with the best estimate value of scenario-specific risk contained in this report.

1.9 Structure of NUREG-1935 and Supporting Documents

The SOARCA project is documented in multiple reports. This volume, NUREG-1935, describes the approach and procedures used in the study and summarizes the project results and conclusions. NUREG/CR-7110, Volumes 1 and 2, contain detailed descriptions of the plant-specific SOARCA analyses and results for the Peach Bottom and Surry plants, respectively. Because this volume and the NUREG/CR reports rely on highly technical explanations, an information brochure (NUREG/BR-0359) was developed as a plain-language summary of SOARCA's methods, results, and conclusions.

An external committee of subject matter experts peer reviewed the SOARCA project in meetings held in July 2009, September 2009, March 2010, November 2010, and December 2011. An appendix to this document will contain the final reports of the peer review committee.

The SOARCA project was nearly at the end of its peer review when the Fukushima Daiichi accident occurred in Japan on March 11, 2011. This accident presented real information regarding the progression of severe accidents and many insights with potential parallels to SOARCA's analysis of SBO scenarios at Peach Bottom, a similarly designed plant. The SOARCA team developed an appendix to this volume which qualitatively compares and contrasts specific accident phenomena based on information available to date. As additional information becomes available, the NRC will continue to review it for lessons learned and insights potentially applicable to nuclear plants in the United States.

2.0 ACCIDENT SCENARIO SELECTION

An accident sequence begins with the occurrence of an initiating event (e.g., a loss of offsite power, a LOCA, or an earthquake) that perturbs the steady-state operation of the NPP. The initiating event challenges the plant's control and safety systems, the failure of which could cause damage to the reactor fuel and result in the release of radioactive fission products. Because an NPP has numerous diverse and redundant safety systems, many different accident sequences are possible, depending on the type of initiating event that occurs, the amount of equipment that fails, and the nature of the operator actions involved.

One way to systematically identify possible accident sequences is to develop accident sequence logic models using event tree analysis, as is done in PRAs. Pathways through an event tree represent accident sequences. Typically, the analysis is divided into two parts: (1) a Level 1 PRA that represents the plant's behavior from the occurrence of an initiating event until core damage occurs and (2) a Level 2 PRA that represents the plant's behavior from the onset of core damage until radiological release occurs. The development of accident sequence logic models requires detailed information about the plant and the expertise of engineers and scientists from a wide variety of technical disciplines. As a result, the construction of accident sequence logic models is a complex and time-consuming activity.

The NRC and NPP licensees have already completed many PRAs. However, because of the improvements in PRA technology and plant capabilities and performance, this study gave more importance to the most current PRA information.

2.1 Approach

Figure 2 illustrates the overall process used to identify and characterize accident scenarios for the SOARCA project. The SOARCA team selected scenarios from the results of existing PRAs. Some of these existing PRAs model accident sequences to the point of radiological release (i.e., they are Level 2 PRAs); however, the majority of existing PRAs are limited to the onset of core damage (i.e., Level 1 PRAs). The team identified core damage sequences from previous staff and licensee PRAs and separated them into core damage groups. A core damage group consists of core damage sequences that have similar timing for important severe accident phenomena and similar containment or engineered safety feature operability. The groups were screened according to their approximate CDFs to identify those that were the most significant. Finally, the accident scenario descriptions were augmented by assessing the status of containment systems (which are not typically modeled in Level 1 PRAs).

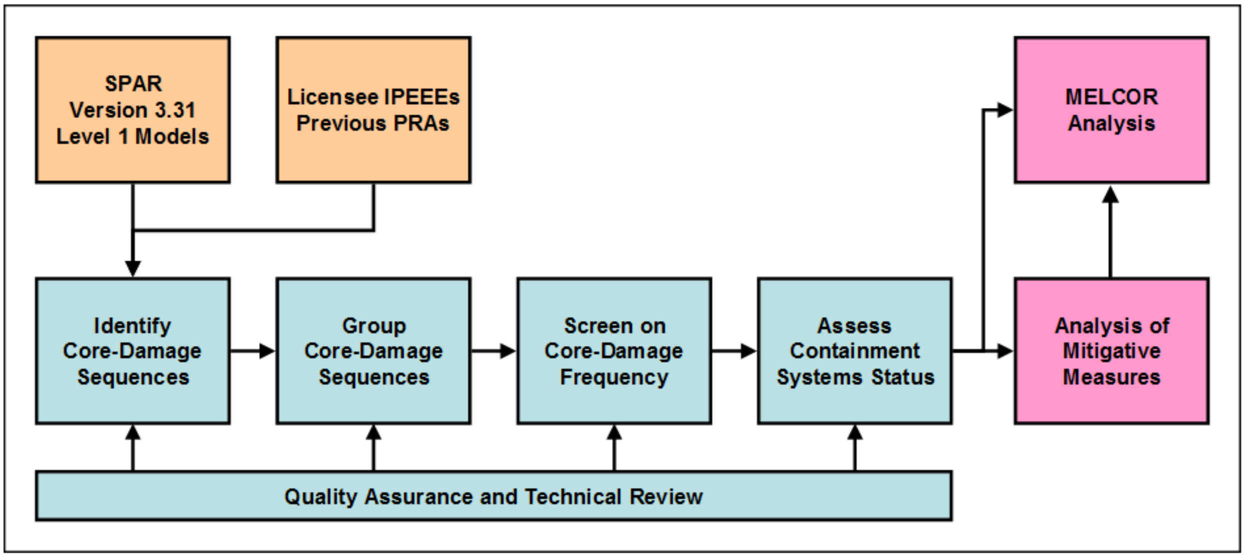

Figure 2 SOARCA accident scenario selection and analysis process

The scope of analyses using MELCOR and MACCS2 was generally confined to scenarios based on the following CDF screening guidelines:

- 10^{-6} per reactor-year for most scenarios

- 10^{-7} per reactor-year for scenarios that are known to have the potential for higher consequences (e.g., containment bypass scenarios such as steam generator tube rupture (SGTR) and ISLOCA initiators)

To accomplish this, the project grouped the release characteristics so that they are representative of scenarios binned into those groups. In addition, the groups are sufficiently broad to include the potentially risk significant but lower frequency scenarios. As a result of limitations in available Level 2 analyses and models, the team selected and screened the scenarios using CDF per reactor-year as the criterion, rather than radionuclide release frequency.

The application of the screening criteria to the available Level 1 PRA information for the pilot plants resulted in the identification of two basic types of scenarios: SBOs and bypass scenarios. This result presents certain advantages with respect to the inherent adequacy of the criteria and of the scope of scenarios. First, SBO scenarios are representative of a broad class of events in PRA—loss of heat removal events. Selection of SBO events in SOARCA ensures that the project covers that broader class of transients involving a loss of heat removal, and further, including an STSBO reasonably bounds the radionuclide release time and consequences of that class of accidents (which could include other events, such as loss of service water or loss of component cooling water but which develop more slowly). Also, for the PWR, the SBO includes, in part, the effect of a small LOCA by considering reactor coolant pump seal leakage. Additionally, selecting SBO sequences for analysis meant including the effects of loss of containment heat removal (fan coolers) and loss of containment spray systems (which are all electrically powered) to remove airborne radionuclides. Thus, the nonbypass sequences also result in containment failure, which would not be the case for all other transients involving such loss of heat removal in a typical PRA. Therefore, while SOARCA used CDF for screening, in effect, the CDF in these cases also represents the radionuclide release frequency.

While the study did not include medium or large loss-of-inventory accidents—because of their very low frequency—it should be noted that such internal events are well below the screening criteria for the BWR and comfortably below the screening criterion for the PWR. For Peach Bottom, the medium and large LOCAs had CDFs of 2×10^{-9} and 1×10^{-9} per reactor-year. For Surry, the medium and large LOCAs had frequencies of 6×10^{-8} and 7×10^{-10} per reactor-year. Only a fraction of these sequences would have resulted in containment failure, because there may not have been a loss of containment heat removal. Since the Surry analyses included an ISLOCA sequence, it can also be argued that they reasonably bounded the radionuclide release time and consequences of events involving a LOCA inside containment for that plant.

The timing of a severe accident's offsite release has a major impact on both early and LCF risks. In this respect, the team examined candidate SOARCA sequences with the timing of both core damage and containment failure in mind. As part of this consideration, it addressed, for the Peach Bottom plant, an additional sequence, the STSBO, even though it fell below the screening criterion. The STSBO frequency is roughly an order of magnitude lower than the LTSBO (3×10^{-7} per reactor-year versus 3×10^{-6} per reactor-year); however, the STSBO has a more prompt radiological release and a slightly larger release over the same interval of time. The initial qualitative assessment of the STSBO concluded that it would not have greater risk significance than the LTSBO, because, while it has a more prompt release (8 hours versus 20 hours), the release is delayed beyond the time needed for successful evacuation. To demonstrate the points regarding risk versus frequency for lower frequency events, the study nonetheless included a detailed analysis of the STSBO. In a related fashion, the study included an ISLOCA sequence for Surry, even though it fell below the screening criterion of 1×10^{-7} per reactor-year for bypass scenarios. Past studies (e.g., NUREG-1150) cited this scenario as important, and it has the potential for larger releases because of its direct release outside the containment.

Finally, the team routinely considered core damage initiators and phenomenological containment failure modes in SOARCA that were considered in the past, except for those that were excluded by extensive research (alpha mode failure, direct containment heating, and gross failure without prior leakage). The detailed analysis includes modeling behavior (including radionuclide transport and release) associated with long-term containment pressurization, Mark I liner failure, induced SGTR, hydrogen combustion, and core concrete interactions.

SOARCA does not include analysis of an extreme earthquake that directly results in a large breach of the RCS (large LOCA), a large breach of the containment, and an immediate loss of safety systems. Given the considerable uncertainties in the quantification of seismic loads and seismic fragilities, in particular the quantification of the size of a hole or the amount of leakage, more research is needed to perform a best estimate analysis. In addition, it would not be sufficient to perform a nuclear plant risk evaluation of this event without also assessing the concomitant nonnuclear risk associated with such a large earthquake. This assessment would have to include an analysis of the impact on public health of an extremely large earthquake—

larger than that generally considered in residential or commercial construction codes—to provide the perspective on the relative risk posed by operation of the plant.

Additionally, SOARCA considered whether the seismic events evaluated for Surry could cause liquefaction-induced settlements large enough to result in containment failure at containment penetrations. A review of previous work related to liquefaction at the Surry site and preliminary analyses assessed the potential for liquefaction-induced soil deformations. According to NUREG/CR-4550 [68], liquefaction is expected to occur for a seismic event greater than the safe-shutdown earthquake (SSE) at Surry. Estimated liquefaction-induced settlements provided in NUREG/CR-4550 range between 2 and 4 inches for a peak ground acceleration of 0.3 to 0.4 g. Using geotechnical data provided in the Surry updated final safety analysis report [69] and the original geotechnical investigation report by Dames and Moore [70], analyses for the SOARCA study resulted in similar settlement estimates in the vicinity of the containment structure, auxiliary building, and turbine building for a peak ground acceleration of 0.4 g. These estimated settlements are considered to be a mean estimate. A site examination performed by engineers for the NUREG/CR-4550 study of the piping systems and cable penetrations going from the auxiliary, safety area, service, and turbine buildings into the containment indicated that such displacements were not likely to cause failure. NUREG/CR-4550 did not provide the basis for this assessment, and this study did not include additional analyses on piping systems to confirm this assessment. Additional settlement analyses were performed for a peak ground acceleration of 0.75 g, which is associated with an event having an annual frequency of occurrence on the order of $1\text{x}10^{-6}$ to $1\text{x}10^{-7}$. At this ground motion level, mean settlement estimates increase to between 4 and 8 inches [71]. The effects of this magnitude of settlement on piping systems have not been assessed in SOARCA. Because of the considerable uncertainties in the quantification of these effects for this magnitude of settlement estimates, more research is needed to perform a best-estimate analysis.

In summary, SOARCA addresses the more likely (though still remote) and important sequences that are understood to compose much of the severe accident reactor risk from nuclear plants. NRC staff conclude that the general methods of SOARCA (i.e., detailed, consistent, phenomenologically based, sequence-specific, accident progression analyses) are applicable to PRA methodology and should be the focus of improvements in that regard.

2.2 Scenarios Initiated by Internal Events

The study identified scenarios initiated by internal events and the availability of containment systems for these scenarios using the NRC's plant-specific SPAR models, licensee PRAs, and NUREG-1150 [2]. The SPAR models support the NRC's oversight of licensed commercial NPPs and have been developed and maintained under a formal quality assurance program. The Peach Bottom SPAR model has been peer reviewed against staff-endorsed industry consensus PRA standards. Both the Surry and Peach Bottom licensee PRAs have been peer reviewed against the same standards. In addition, the SPAR model accident sequence results (including the sequence minimal cut sets) are periodically compared to the results from licensee PRAs under the Mitigating System Performance Index Program, which is part of the NRC's Reactor Oversight Process. As a result, both the qualitative and quantitative results from the Surry and

Peach Bottom SPAR models are in reasonable agreement with the corresponding licensee PRAs. Specific comparisons are discussed below.

The following process determined the scenarios for further SOARCA analyses:

- Candidate accident scenarios were identified in analyses using plant-specific SPAR models (Version 3.31).
  - Initial Screening. Screened-out sequences with a CDF less than 10^{-8}, eliminating 4 percent of the overall CDF for Peach Bottom and 7 percent of the overall CDF for Surry.
  - Sequence Evaluation. Identified and evaluated the dominant cutsets for the remaining sequences. Determined system and equipment availabilities and accident sequence timing.
  - Scenario Grouping. Grouped sequences with similar times to core damage and equipment availabilities into scenarios.
- Containment systems availabilities for each scenario were assessed using system dependency tables that delineate the support systems required for performance of the target front-line systems and from a review of existing SPAR model system fault trees.
- Core damage sequences from the licensee PRA model were reviewed and compared with the scenarios determined by using the SPAR models. Differences were resolved during meetings with licensee staff.
- The screening criteria (CDF less than 10^{-6} for most scenarios and less than 10^{-7} for containment bypass sequences) were applied to eliminate scenarios from further analyses.

This process provides the basic characteristics of each scenario. However, it is necessary to have more detailed information about each scenario than is contained in a PRA model. Capturing the additional scenario details requires further analysis of system descriptions and a review of procedures. This review includes the analysis of mitigation measures beyond those treated in current PRA models. Mitigation measures treated in SOARCA include the plant-specific EOPs, SAMGs, and 10 CFR 50.54(hh) mitigation measures. Section 0 describes the mitigation measures assessment process used to determine what measures would be available and the associated timing to implement them.

2.3 Scenarios Initiated by External Events

As explained in Section 2.1, the SOARCA team considered and selected accident scenarios (sequence groups, rather than individual sequences) based on both likelihood and potential consequences. The team identified core damage sequences from previous staff and licensee PRAs and separated them into core damage groups. It then screened the groups (not individual sequences) according to their approximate CDFs to identify the most significant ones. Since core damage groups (i.e., scenarios) were considered, many individual lower order sequences would be captured in the aggregation into groups.

External events include internal flooding and fire; seismic events; extreme wind-, tornado-, and hurricane-related events; and similar events that may apply to a specific site. The external event scenarios developed for SOARCA analysis were derived from a review of past studies, such as the NUREG-1150 study [2], IPEEE submittals, and other relevant generic information. Detailed sequence characteristics are more difficult to specify for external event scenarios because of the general lack of external event PRA models industrywide. As a result, the SOARCA external event scenarios are heuristically based (i.e., experience based), as opposed to the internal event scenarios, which were developed through more formal, rigorous PRA methods.

These scenarios were initiated by a seismic, fire, or flooding event. The mitigation measures assessment for each of these scenarios assumed that the initiator was a seismic event, because it was judged to be limiting. Seismic initiators are considered to be limiting for two principal reasons. First, they are more likely to result in the near immediate failure of systems, whereas, fire and flood would be expected to result in delayed failures. Secondly, a seismic event may be more likely than a fire or flood to fail passive components, such as water tanks. Additionally, seismic initiators may be more likely to have sitewide and offsite impacts.

No attempt was made to match the frequencies of the external event scenarios to the actual sequence frequencies in any of the input information sources, because much of the available quantitative risk information on external events is dated. For example, since the publication of input information sources, new seismic hazard estimates have been developed. As a result, the estimated frequencies of the external event scenarios were based on expert judgment that considered the impact of changes in seismic data and methods on the published external-event PRA results. Care was taken to ensure that the external event scenario selection maintained the relative importance of external events CDF versus internal events CDF.

2.4 Accident Scenarios Selected for Surry

The SOARCA team selected four accident scenarios for the Surry plant (two initiated by internal events and two initiated by external events). The following sections identify each selected accident scenario, provide its representative CDF, and summarize the accident scenario in terms of its initiating event, equipment failures, and operator errors.

2.4.1 Surry Internal Event Scenarios

Two internal event scenarios for Surry met the criteria for further analysis.

(1) Initiating Event: Spontaneous SGTR

Representative CDF: 5×10^{-7} per reactor-year (SPAR)

Scenario Summary: This scenario is initiated by a spontaneous rupture in one steam generator tube. The operators fail to (1) isolate the faulted steam generator, (2) depressurize and cool down the RCS, and (3) refill the refueling water storage tank

(RWST) or cross-connect to the unaffected unit's RWST. Auxiliary feedwater, high-pressure injection, low-pressure injection, and containment spray are available, if needed. However, high-pressure recirculation, low-pressure recirculation, and the recirculation sprays will be unavailable as a result of lack of water in the containment sump.

Comparison with Licensee PRA: The licensee PRA calculates a CDF of 1×10^{-6} per reactor-year for this scenario. The conditional core damage probabilities are virtually identical for the SPAR analysis (1.4×10^{-4}) and for the licensee PRA (1.5×10^{-4}). The difference in the calculated CDFs is mainly attributable to the difference in initiating event frequency. Because both the SPAR model and licensee-calculated CDFs for this scenario are above the 1×10^{-7} per reactor-year threshold for containment bypass scenarios, this scenario was retained for further analysis.

(2) Initiating Event: ISLOCA in the Low-Head Safety Injection System

Representative Frequency: $3\times10e^{-8}$ per reactor-year (SPAR)

Scenario Summary: This scenario is initiated by a common-cause failure of both low-head safety injection (LHSI) inboard isolation check valves. The open pathway pressurizes and ruptures a section of the low-pressure piping outside the containment, which opens a containment bypass LOCA. This sequence group consists of the bypass LOCA, followed by operator failures to refill the RWST or cross-connect to the unaffected unit's RWST. The ability to inject using the LHSI is not possible because of the pipe rupture. The high-head injection system remains available, because the pumps are in a separate location. Core damage occurs because of RWST depletion and operator failure to refill the RWST or cross-connect to the unaffected unit's RWST.

Comparison with Licensee PRA: The ISLOCA scenario analyzed in SOARCA is a catastrophic failure of both of the inboard isolation check valve disks within the LHSI piping, together with failure to refill the RWST or to cross-connect to the unaffected unit's RWST. For this ISLOCA scenario, the NRC's SPAR model calculated a CDF of $3x10^{-8}$ per reactor-year, and the NRC's initial understanding was that the licensee's PRA calculated a CDF of $7x10^{-7}$ per reactor-year. SOARCA analyses originally included this scenario because the licensee's PRA for Surry included an ISLOCA frequency of $7x10^{-7}$ per reactor-year, and it has been commonly identified as an important contributor in PRA.

During Surry site visits on January 19, 2011, and October 26, 2011, the NRC staff learned that the licensee's current PRA model has the following two ISLOCA scenarios:

- scenario one: catastrophic failure of one check valve, leak-by of the second check valve, and the motor-operated isolation valve being unable to close

- scenario two: catastrophic failure of two check valves

Scenario one would result in a leak between 50–300 gallons per minute (gpm) from the RCS. Anything less than 50 gpm would be mitigated by a relief valve on the low pressure side of the LHSI injection line; pipe rupture would not occur. The frequency of the catastrophic failure of one check valve and the leak-by of the second check valve is 1×10^{-6} per reactor-year. When compounded by all the potential failure modes (including operator error and mechanical or electrical failures) of the motor-operated valve, that lowers the frequency of scenario one to 7×10^{-7} per reactor-year. This frequency does not include any consideration of averting core damage by refilling or cross-connecting RWSTs. This is a significant conservatism.

Scenario two would result in a leak above 300 gpm from the RCS. The licensee‘s current PRA model assumes that the probability for the catastrophic failure of both isolation check valves is approximately 3×10^{-8} per reactor-year. As with scenario one, this frequency does not include consideration of averting core damage by refilling or cross-connecting RWSTs. Scenario two does not meet the SOARCA screening criterion of 1×10^{-7} per reactor-year for a bypass event. However, the team elected to retain it, because it has been commonly identified as an important contributor in PRA.

2.4.2 Surry External Event Scenarios

Two external event scenarios for Surry met the criteria for further analysis.

(1) Initiating Event: Seismic-initiated LTSBO

Representative Frequency: 1×10^{-5} to 2×10^{-5} per reactor-year

Scenario Summary: This scenario is initiated by an earthquake of 0.3–0.5 g peak ground acceleration (PGA). The seismic event results in loss of offsite power (LOOP) and failure of onsite emergency alternating current (ac) power, resulting in an SBO event where neither onsite nor offsite ac power are recoverable. All systems dependent on ac power are unavailable, including the containment systems (containment spray and fan coolers). The TDAFW system is available initially. Eventually, loss of the TDAFW occurs because of battery depletion and the resulting loss of direct current (dc) power for sensing and control. The loss of pump seal cooling will cause a reactor coolant pump seal to leak.

(2) Initiating Event: Seismic-Initiated STSBO

Representative Frequency: 1×10^{-6} to 2×10^{-6} per reactor-year

Scenario Summary: This scenario is initiated by an earthquake of 0.5–1.0 g PGA. The seismic event results in a LOOP and failure of onsite emergency ac power, resulting in an SBO event where neither onsite nor offsite ac power are recoverable. All systems dependent on ac power are unavailable, including the containment systems (containment spray and fan coolers). The seismic event also results in a loss of dc power, resulting in the loss of automatic control of the TDAFW system. The earthquake ruptures the emergency condensate storage tank (ECST), which is conservatively assumed to empty immediately, rendering the TDAFW system initially unavailable. This scenario is referred to as the STSBO, since the site loses all power, even the batteries, and therefore all of the safety systems become quickly inoperable in the "short term."

(3) Initiating Event: Seismic-Initiated STSBO with Induced SGTR

Representative Frequency: 3×10^{-7} to 5×10^{-7} per reactor-year. The representative frequency for this event is estimated to be 3.75×10^{-7} per reactor-year, based on an assumed conditional tube failure probability of 0.25, selected from NUREG-1570, "Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture," issued March 1998 [7].

Scenario Summary: An additional seismic-initiated STSBO scenario involves a variation that considers the conditional likelihood of a thermally induced steam generator tube rupture (TISGTR).

2.5 Accident Scenarios Selected for Peach Bottom

The SOARCA team selected two accident scenarios for the Peach Bottom plant (both initiated by a seismic event). In addition, SOARCA included a mitigation assessment for the Loss of Vital AC Bus E-12 scenario. The following sections identify each selected accident scenario, provide its representative CDF, and summarize the scenario in terms of its initiating event, equipment failures, and operator errors.

2.5.1 Peach Bottom Internal Event Scenarios

The Loss of Vital AC Bus E-12 was initially estimated to have a frequency above the SOARCA screening criterion of 1×10^{-6} per reactor-year and was therefore analyzed. However, after further review of the SPAR model and comparison with the licensee's PRA, the team determined that the scenario had a CDF below the screening criteria. Because the MELCOR analysis provided unique insights into the mitigation and response of the plant for this internal event sequence, the team retained the MELCOR analysis.

2.5.2 Peach Bottom External Event Scenarios

(1) Initiating Event: Seismic-Initiated LTSBO

Representative Frequency: 1×10^{-6} to 5×10^{-6} per reactor-year

Scenario Summary: This scenario is initiated by an earthquake of 0.3–0.5 g PGA. The seismic event results in a LOOP, failure of onsite emergency ac power, and failure of the Conowingo Dam power line, resulting in an SBO event where neither onsite nor offsite ac power are recoverable. All systems dependent on ac power are unavailable, including the containment systems (containment spray). The turbine-driven injection systems—high-pressure coolant injection (HPCI) or RCIC, or both—are available until battery depletion.

(2) Initiating Event: Seismic-Initiated STSBO

Representative Frequency: 1×10^{-7} to 5×10^{-7} per reactor-year

Scenario Summary: This scenario is initiated by an earthquake of 0.5–1.0 g PGA. The seismic event results in a LOOP, failure of onsite emergency ac power, and failure of the Conowingo Dam power line, resulting in an SBO event where neither onsite nor offsite ac power are recoverable. All systems dependent on ac power are unavailable, including the containment systems (containment spray). In addition, HPCI and RCIC are initially assumed to be unavailable because of the loss of dc power. The larger earthquake ruptures the condensate storage tank (CST). The earthquake causes the fire water system to fail. This scenario is referred to as the short-term SBO since the site loses all power, even the batteries, and therefore all of the safety systems become quickly inoperable in the "short term."

Note: The STSBO scenario does not meet the SOARCA screening criterion of 1×10^{-6} per reactor-year; however, the team retained the scenario for analysis to assess the risk importance of a lower frequency, potentially higher consequence scenario. This type of scenario has been a risk-important severe accident scenario in past PRA studies. The SOARCA study analyzed two variations of this scenario, one with and one without RCIC blackstart.

2.6 Generic Factors

The results of existing PRAs indicate that the likelihood of an NPP accident sequence that releases a significant amount of radioactivity is very small, owing to the diverse and redundant barriers and numerous safety systems in the plant, the training and skills of the reactor operators, testing and maintenance activities, and the regulatory requirements and oversight of the NRC. In addition, it is important to recognize that CDFs of NPPs have decreased over the years. Several reasons exist for these decreases:

- Utilities have completed plant modifications intended to remedy concerns raised in earlier PRAs.

- Plants exhibit better performance as evidenced by reductions in initiating event frequencies, improvements in equipment reliability, and higher equipment availability. NPP equipment has become more reliable and available because of improved maintenance practices motivated by the Maintenance Rule (10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants") [8].

- The NRC has issued new regulations, such as the Anticipated Transient without Scram (ATWS) Rule (10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants") [9] and the SBO Rule (10 CFR 50.63, "Loss of All Alternating Current Power") [10] that directly affect the likelihood of certain types of accidents. Although the NRC issued the ATWS Rule and the SBO Rule before it completed NUREG-1150 [2], it did not address the impact of these rules on risk in NUREG-1150.

- PRA methodologies have improved, allowing a more realistic assessment of risk. In this category, improvements in common-cause failures analysis are noteworthy.

As a result, risk estimates reflect the impacts of constantly changing plant operational, regulatory, and PRA technology environments. Any attempt to identify significant accident sequences should be viewed as a "snapshot" of the plant at the time the analysis was completed.

3.0 MITIGATIVE MEASURES ASSESSMENT

The overall objective of the SOARCA project is to develop a body of knowledge regarding the realistic outcomes of severe reactor accidents. Included within this objective is to provide insight into the effectiveness and benefits of mitigation measures currently employed at operating reactors. Section 2.0 describes the PRA information sources, including the NRC's SPAR models, licensees' PRA models, NUREG-1150, and additional expert judgment that this study used to identify risk-important sequence groups leading to core damage and containment failure or bypass. This section describes the methods that determined the mitigation measures that would be available and the associated timing to implement them. This includes mitigation measures beyond those treated in current PRA models. Mitigation measures treated in SOARCA include the licensee's EOPs, SAMGs, and 10 CFR 50.54(hh) mitigation measures. It is expected that the licensee's emergency response organization would implement these measures in accordance with the approved emergency plan.

3.1 Site-Specific Mitigation Strategies

In preparation for the detailed, realistic modeling of accident progression and offsite consequences, the SOARCA project staff had extensive cooperation from the licensees to develop high-fidelity plant systems models; define operator actions, including the most recently developed mitigative actions; and develop models to simulate site-specific and scenario-specific emergency planning. In addition to input for model development, licensees provided information from their own PRAs on accident scenarios. Through tabletop exercises (with senior reactor operators, PRA analysts, and other licensee staff) of the selected scenarios, licensees provided input on the timing and nature of the operator actions to mitigate the selected scenarios. The licensee input for each scenario was used to develop timelines of operator actions and equipment lineup or setup times for implementing the available mitigation measures. This includes mitigation measures beyond those treated in current PRA models.

The SOARCA team developed the timelines for implementing the mitigation measures directed in plant-specific procedures and mobilizing support organizations after discussing each scenario with licensee personnel who have experience in operations, engineering, and facility management. The team developed these timelines through multiple site visits and system walkdowns in 2007, 2010, and 2011, with licensee personnel specifically reviewing the steps to implement mitigation. Results of preliminary accident progression calculations were used to characterize anticipated changes in plant conditions and describe the signatures of measurable parameters. Estimates were then made for the time needed to assemble necessary personnel, tools, and equipment; align and start components; and establish a desired operating condition. For the ISLOCA scenario, where the timing of operator actions was judged to be important to the results, the licensee performed plant simulator runs with reactor operators to ensure that the timing for key actions was as realistic as possible.

Mitigation measures treated in SOARCA include EOPs, SAMGs, and 10 CFR 50.54(hh) mitigation measures. The 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. NRC inspectors completed the verification of licensee implementation (i.e., equipment, procedures, and training) of 10 CFR 50.54(hh) mitigation measures in December 2008. These mitigation measures are for use during scenarios involving large fires and explosions. One such measure is portable, self-powered equipment, including generators and diesel-driven pumps. Portable generators provide electrical power to equipment that gives critical indications, such as the reactor vessel water level. Portable generators also provide electrical power needed to operate safety relief valves. Portable diesel-driven pumps provide a diverse and independent means of injecting water into the RCS and steam generators. Another such measure is starting and controlling, without electrical control power, the plant's existing turbine-driven injection systems, including the RCIC and TDAFW systems.

To quantify the benefits of the mitigation measures and to provide a basis for comparison to past analyses of unmitigated severe accident scenarios, the project team also analyzed the scenarios assuming that the events proceed as unmitigated by key available onsite mitigation measures, ultimately leading to core damage and an offsite release. This NUREG refers to these as "unmitigated scenarios," because they are not effectively mitigated by onsite resources.

3.1.1 Scenarios Initiated by External Events

Scenarios identified in SOARCA included both externally and internally initiated events. The externally initiated events included events for which seismic, fire, extreme wind, and flooding initiators were grouped together.

The PRA screening identified the following scenarios that were initiated by external seismic, fire, or flooding events:

- Peach Bottom LTSBO: 1×10^{-6} to 5×10^{-6}/reactor-year
- Surry LTSBO: 1×10^{-5} to 2×10^{-5}/reactor-year
- Surry STSBO: 1×10^{-6} to 2×10^{-6}/reactor-year
- Surry STSBO with TISGTR: 3×10^{-7} to 5×10^{-7}/reactor-year

The mitigation measures assessment for each of these scenarios assumed that the initiator was a seismic event, because it was judged to be limiting. Seismic initiators are considered to be limiting for two principal reasons. First, seismic initiators are more likely to result in the near immediate failure of systems, whereas, fire and flood would be expected to result in delayed failures. Secondly, a seismic event may be more likely than a fire or flood to fail passive components, such as water tanks. Additionally, seismic initiators may be more likely to have sitewide and offsite impacts.

It is important to note that, although it is not included in the above list, the seismically induced Peach Bottom STSBO was also retained for analysis. With a frequency of 1×10^{-7} to 5×10^{-7}/reactor-year, this scenario does not explicitly meet the SOARCA screening criterion. Nonetheless, it was retained to assess the risk importance of a lower frequency, potentially higher consequence scenario. The STSBO has also been an important event in many past PRAs and is limiting in many transients.

Seismic events considered in SOARCA result in loss of offsite and onsite ac power and, for the more severe seismic events, loss of dc power. Under these conditions, the turbine-driven RCIC and TDAFW systems are important mitigation measures. BWR SAMGs include starting RCIC without electricity to cope with SBO conditions. This is known as RCIC blackstart. The 10 CFR 50.54(hh) mitigation measures have taken this a step further and also include long-term operation of RCIC without electricity (RCIC blackrun), using a portable generator to supply power to indications, such as the RPV level indication, to allow the operator to manually adjust RCIC flow to prevent RPV overfill and flooding of the RCIC turbine. Similar procedures have been developed for PWRs for TDAFW. For the Peach Bottom and Surry LTSBO scenarios, RCIC and TDAFW can be used to cool the core until battery exhaustion. In addition, blackstart procedures can be used for the Peach Bottom STSBO scenario. After battery exhaustion, blackrun of RCIC and TDAFW systems can continue to cool the core. The study used MELCOR calculations to demonstrate core cooling under these conditions.

Seismic PRAs for Peach Bottom and Surry do not describe general plant damage and accessibility. The damage was assumed to be widespread and accessibility to be difficult, consistent with the unavailability of many plant systems.

The seismic initiating event for the SBO accident scenarios might rupture the CST, which is the primary water reservoir for RCIC. However, the Peach Bottom CST is surrounded by a reinforced concrete dike or moat, which would retain water drained from the CST. Therefore, suction from the CST would not be interrupted by a loss of CST integrity. Makeup to the CST would likely be available from the cooling water tower basin (3.55 million gallons or 13,438 cubic meters), or the Susquehanna River; the diesel-driven portable pump (i.e., 10 CFR 50.54(hh) equipment) or other mobile equipment could be used.

For the Surry LTSBO, the TDAFW pump is available until the ECST empties. The ECST initially supplies the TDAFW pump but has finite resources (i.e., it empties in 5 hours). However, the team estimated that the operators would have sufficient time, access, and resources to make up water for injection into the ECST. The low-pressure injection and safety-related containment spray piping were judged not likely to fail for this scenario. The integrity of this piping provided a connection point for a portable, diesel-driven pump to inject into the RCS. Licensee staff estimated that transporting the pump and connecting it to plant piping would take about 2 hours, leading to vessel injection at 3.5 hours, or 2 hours after the operators and support staff recommended the action. Consequently, the cooling water would be supplied to the steam generators for RCS heat removal. The team assumed that operators would eventually provide makeup water to the ECST.

One Surry STSBO assumption was that the ECST would fail and an alternative reservoir would be available within 8 hours; using a fire truck or portable pump to draw from the discharge canal. The low-pressure injection and containment spray safety-related piping were judged not likely to fail, based primarily on NUREG/CR-4334, "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants," issued in 1985 [11], to help extrapolate the potential viability of safety-related piping after a 1.0 g event. This conclusion also considered related studies, including a 2007 German study, "Seismic PSA of the Neckarwestheim 1 Nuclear Power

Plant" [12], that physically simulated ground motion equal to 1 g on an existing plant. The integrity of this piping provided a connection point for a portable, diesel-driven pump to inject into the RCS or into the containment spray systems. Licensee staff estimated that transporting the pump and connecting it to plant piping would take about 2 hours. However, for the STSBO, this mitigation measure was conservatively estimated to take 8 hours, owing to the higher level of damage. Because the installation time was beyond the estimated time to fuel damage and vessel failure (3 hours to core damage, 7 hours to lower head failure), the containment spray system was the preferred mitigation measure. A better understanding of the effect of large seismic events on general plant conditions would be helpful in reducing the uncertainty in availability and accessibility for mitigation measures. If accessibility was not significantly impaired and delay in using the portable pump was limited to 2 hours, then core damage could be averted.

The 10 CFR 50.54(hh) mitigation measures include portable equipment (such as portable power supplies to supply indication, portable diesel-driven pumps, and portable air bottles to open air-operated valves), together with procedures to implement these measures under severe accident conditions. Surry's portable equipment and fire truck are stored onsite in a one-story, multibay garage. Some of Peach Bottom's portable equipment is stored in an open bay in the water treatment building and some is stored outside under a tarp.

The SOARCA team estimated the time to implement individual mitigation measures based on licensee input for each scenario; these estimates take into account the plant conditions following the seismic event. Also, for portable equipment, the time estimates reflect exercises run by licensee staff that provided actual times to move the equipment into place and were adjusted (increased) to account for the larger seismic event. The time estimates for staffing the TSCs and the emergency operating facilities (EOFs) were based on regulatory requirements and the potential for additional delays resulting from the possible effect of the seismic event on roads and bridges.

The mitigation measures assessment noted the possibility of bringing in offsite equipment (e.g., fire trucks, pumps, and power supplies from sister plants or from contractors), but it did not quantify the types, amounts, and timing of this equipment arriving and being implemented. This equipment is also judged to be effective in mitigating an environmental release (by flooding core debris) after it begins. Section 3.2 provides additional information on equipment available offsite and time estimates for transporting this equipment.

Because the SOARCA project did not analyze multiunit accidents, the mitigation measures assessment for external events assumed that the operators only had to mitigate an accident at one reactor, even though Peach Bottom and Surry are two-unit sites. It also assumed minimum staffing and that half of the onsite operators mitigate the damaged unit. Peach Bottom had voluntarily arranged to provide redundant 10 CFR 50.54(hh) equipment to mitigate both units simultaneously; however, SOARCA did not examine this.

3.1.2 Scenarios Initiated by Internal Events

The PRA screening identified the following scenarios that were initiated by internal events:

- Surry interfacing systems loss-of-cooling accident (ISLOCA): 3×10^{-8}/reactor-year
- Surry spontaneous SGTR: 5×10^{-7}/reactor-year

These scenarios result in core damage as a result of assumed operator errors. For the ISLOCA, the operators fail to refill the RWST or cross-connect to the other unit's RWST. For the spontaneous SGTR, the operators fail to (1) isolate the faulted steam generator, (2) depressurize and cool down the RCS, and (3) refill the RWST or cross-connect to the unaffected unit's RWST.

The SPAR model and the licensee's PRA concluded that these two events proceed to core damage as a result of the above-postulated operator errors. However, these PRA models do not appear to have credited the significant time available for the operators to correctly respond to events. They also do not appear to credit technical assistance from the TSC and the EOF. For the ISLOCA, the realistic analysis of thermal-hydraulics presented in NUREG/CR-7110, Volume 2, estimated 6 hours until the RWST is empty and 13 hours until fission product release begins, providing time for the operators to correctly respond. The ISLOCA time estimates are based on a double-ended pipe rupture. These estimates would be longer for smaller break sizes. Also, if the operators throttle high-head safety injection to match decay heat, the time to empty the RWST and the beginning of core damage would be extended by an additional 24 hours. For the SGTR, the realistic analysis of thermal-hydraulics showed from 24 to 48 hours until core damage begins. Therefore, based on realistic time estimates by which the technical assistance is received from the TSC and the EOF, it was highly likely that the operators would correctly respond to the events. These time estimates considered indications that the operators would have of the bypass accident, operator training on plant procedures for dealing with bypass accidents and related drills, and assistance from the TSC and EOF, which were estimated to be fully staffed and operational by 1 to 1.5 hours into the event.

The mitigation measures assessment for internal events also included 10 CFR 50.54(hh) mitigation measures, but these were subsequently shown to be redundant to the wide variety of equipment and indications available for mitigating the ISLOCA and SGTR. ISLOCA and SGTR are internal events that involve few equipment failures and are controlled by operator errors.

The PRA screening for Peach Bottom initially identified the Loss of Vital ac Bus E12 scenario as exceeding the SOARCA screening criterion of 1×10^{-6}/reactor-year. However, a simplifying modeling assumption was subsequently found in the SPAR model, and the scenario frequency was determined to be below the SOARCA screening criterion. By the time the issue was discovered, the mitigation measures assessment and the MELCOR analysis were complete. The MELCOR analysis described in NUREG/CR-7110, Volume 1, demonstrated that this scenario did not result in core damage, even without crediting 10 CFR 50.54(hh) mitigation measures, contrary to the more conservative treatment in SPAR. Nevertheless, this report describes the mitigation measures assessment and the MELCOR analysis for this scenario to demonstrate the benefit of a detailed review of success criteria using integrated thermal-hydraulic analysis.

3.2 Truncation of Releases

Many resources at the State, regional, and national level would be available to mitigate an NPP accident. The staff reviewed available resources and emergency plans and determined that adequate mitigation measures could be brought onsite within 24 hours and be connected and functioning within 48 hours.

Concurrent with the NRC and industry response, the National Response Framework (NRF) would establish a coordinated response of national assets. As described in the Nuclear/Radiological Incident Annex to the NRF, the NRC is typically the Coordinating Agency for incidents occurring at NRC-licensed facilities. As Coordinating Agency, the NRC has technical leadership for the Federal Government's response to the incident. Under an established agreement with the NRC, the U.S. Department of Homeland Security would be the Coordinating Agency for an event in which a general emergency is declared. The NRF is exercised periodically and provides access to the full resources of the Federal Government. The NRC has an extensive, well-trained, and exercised emergency response capability and has onsite resident inspectors. These onsite inspectors are equipped and available to provide firsthand knowledge of accident conditions. The NRC would activate the incident response team at the NRC regional office and Headquarters. The focus of the NRC response is to ensure that public health and safety are protected and to assist the licensee with the response by working with the Department of Homeland Security to coordinate the national response. Concurrently, the NRC regional office would send a site team to staff positions in the reactor control room, TSC, and EOF to support the response. The EOF and TSC are assumed manned and operational in roughly 1–2 hours, depending on the accident scenario. The NRC performs an independent assessment of the actions taken or proposed by the licensee to confirm that such actions will arrest the accident.

Both Surry and Peach Bottom are supported by an offsite EOF. The emergency response organization at the EOF has access to fleetwide emergency response personnel and equipment, including the 10 CFR 50.54(hh) mitigation measures and equipment from sister plants. These assets, as well as those from neighboring utilities and State preparedness programs, could be brought to bear on the accident if needed. Every licensee participates in full onsite and offsite exercises every 2 years where response to severe accidents and coordination with offsite response organizations (OROs) is demonstrated and inspected by the NRC and the Federal Emergency Management Agency. In addition, the Institute for Nuclear Power Operations and the Nuclear Energy Institute would activate their emergency response centers to assist the site as needed.

All of the described resources would be available to the site to mitigate the accident. Although some of these efforts would be ad hoc, knowledgeable personnel and an extensive array of equipment would be available and were considered in the conclusion that radiological releases would be truncated within 48 hours except for the Surry LTSBO sequence, which was truncated at 72 hours.

4.0 SOURCE TERM ANALYSIS

Section 4.1 describes some background in key studies for regulatory and probabilistic applications. Figure 3 shows a timeline of key events and NRC studies in the evolution of nuclear safety technology, as well as the key source term studies cited in the timeline that preceded the SOARCA program (also discussed in Section 4.1 below). Section 4.2 contains a history of the severe accident source term codes developed by the NRC and the scope of the MELCOR code. The MELCOR code is the culmination of the NRC research and code development of severe accident phenomena for source term evaluations. Section 4.3 presents the MELCOR modeling approach used in the SOARCA analyses. This includes the development of the plant models, the best practices approaches to important but uncertain phenomena and equipment performance, recent advances in source term models, and the methods used to calculate the radionuclide inventories.

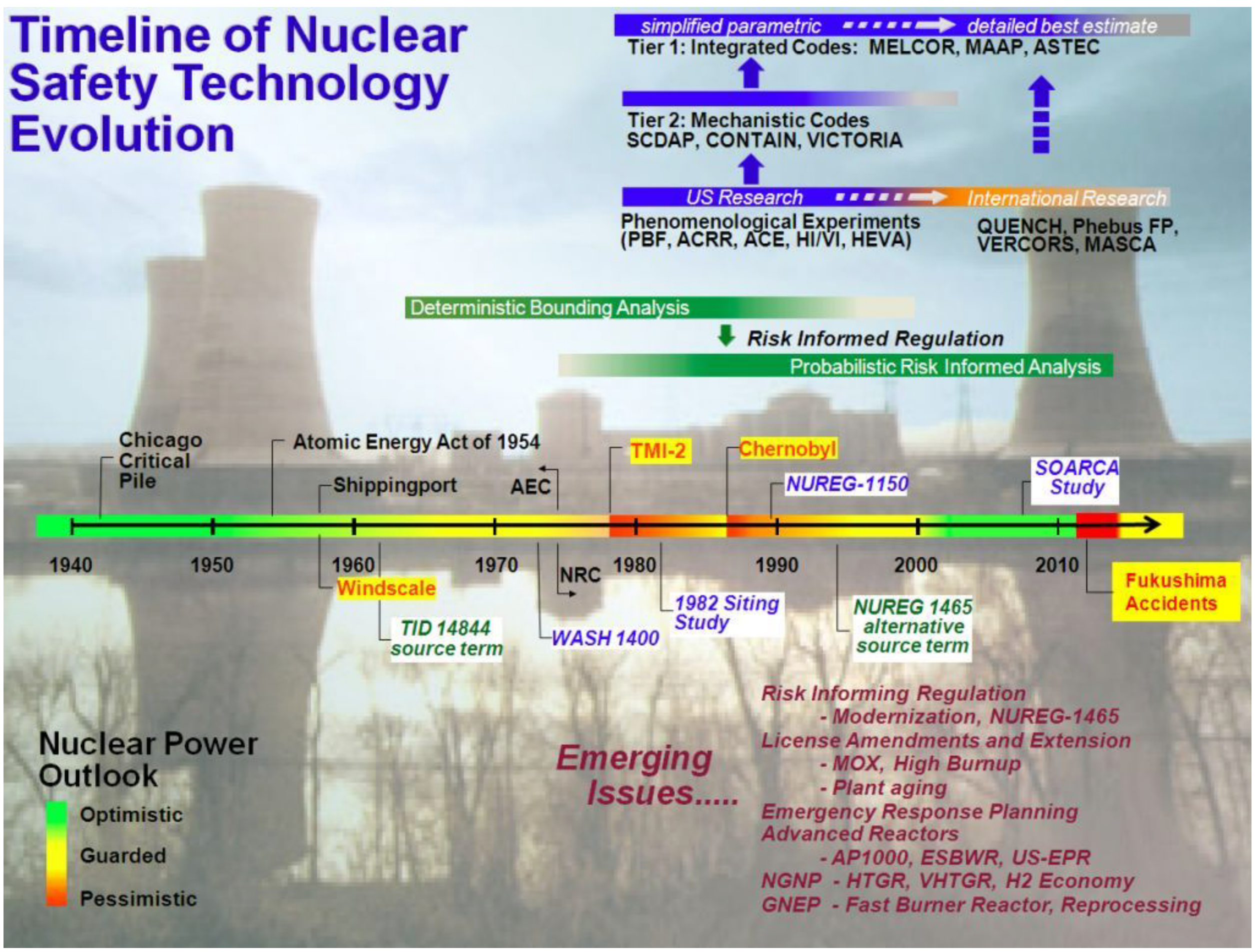

Figure 3 Timeline of key nuclear power events and safety studies

4.1 <u>Source Term Study Background</u>

The Reactor Safety Study (WASH-1400) [3], was the first systematic attempt to provide realistic estimates of public risk from potential accidents in commercial NPPs. The 1975 study included analytical methods for determining both the probabilities and consequences of various accident scenarios. The study used event trees and fault trees to define important accident sequences and to quantify the reliability of engineered safety systems and contained a list of nine PWR and five BWR source terms. All the accidents that were believed to contribute significantly to the overall core melt frequency were grouped, or “binned,” into the source term categories. The WASH-1400 source terms included characterizations of accident timing, the release duration (e.g., puff or sustained release), and the energy of the release for plume loft considerations. The description of radioactivity used eight chemical categories. The 54 most health-significant isotopes were used in health consequence calculations.

The WASH-1400 methodology used to predict the health effects from the source term was based on the newly developed Calculation of Reactor Accident Consequences (CRAC) code [18] that calculated the atmospheric dispersion and health consequences. However, an integrated tool for the calculation of the source term did not exist. The estimation of the source term used the best analytic procedures available at the time. When ample data were available, a model for the phenomenon was included as realistically as possible, but when data were lacking, consideration of the phenomenon was omitted. The resultant source terms reflected uncertainties and poor understanding of applicable phenomena. Uncertainties in accident frequencies were accounted for by adding 10 percent of the likelihood of each release category into the next larger and the next smaller category.

Subsequently, the NRC documented the technical basis for source terms in NUREG-0772, “Technical Bases for Estimating Fission Product Behavior during LWR Accidents,” issued June 1981 [19]. NUREG-0772 assessed the assumptions, procedures, and available data for predicting fission product behavior. Four conclusions of the NUREG-0772 study were (1) a new definition of the chemical form of iodine (I) (i.e., cesium iodide (CsI) was the dominant form), (2) the potential retention of CsI within the vessel or containment versus elemental iodine, (3) the inclusion of in-vessel retention, and (4) the role of containment engineering safety features (e.g., sprays, suppression pools, and ice condensers). However, NUREG-0772 based much of the quantitative assessment on scoping calculations that were applicable only to specific conditions. In particular, it conducted the examination of fission product behavior in different regions of the plant with different accidents in parallel with limited consideration of integral effects. The NRC examined the potential impact of the NUREG-0772 findings on reactor regulation and documented the results in draft NUREG-0771, “Regulatory Impact of Nuclear Reactor Accident Source Term Assumptions,” issued June 1981 [20].

The NUREG-0771 and NUREG-0772 studies formed the basis for the designation of five accident groups as representative of the spectrum of potential accident conditions documented in NUREG-0773, “The Development of Severe Accident Source Terms: 1957–1981,” issued November 1982 [21]. In 1982, the NRC issued the NUREG/CR-2239 siting study [1] using the NUREG-0773 source terms. It determined that the five source terms adequately spanned the range of possible source terms. The source terms, developed from separate effects computer code analyses that were performed in 1978, were used to calculate accident consequences at

91 U.S. reactor sites using site-specific population data and a mixture of site-specific and regionally specific meteorological data. An objective of the SOARCA study is to update this study.

In response to emerging regulatory needs, Battelle Columbus Laboratories conducted a study, "Radionuclide Release Under Specific LWR Accident Conditions," published in 1985 [22], that developed and modified a number of separate effects severe accident computer codes based on emerging severe accident research. The codes, coupled together to form a code suite, could calculate a complete accident sequence. The new Source Term Code Package (STCP) code [22] calculated the source terms for about 25 specific sequences for five operating plants. Although the STCP was a significant step forward in deterministic severe accident analysis, the code suite had some significant shortcomings. Because the code represented the linkage of many separate code modules, the data transfer and feedback effects were not always handled consistently. The technical basis for the models in the STCP is in NUREG-0956, "Reassessment of the Technical Bases for Estimating Source Terms," issued July 1986 [16]. The results from the STCP calculations supported the NUREG-1150 PRA [2], along with expert judgment and simplified algorithms for sequence-specific source terms.

The NUREG-1150 PRA was an effort to put the insights gained from the research on system behavior and phenomenological aspects of severe accidents into a risk perspective. An important characteristic of this study was the inclusion of the uncertainties in the calculations of CDF and source term caused by an incomplete understanding of reactor systems and severe accident phenomena at that time. NUREG-1150 therefore used sensitivity studies, uncertainty studies, and expert judgment to characterize the likelihood of alternative events that affect the course of an accident. The elicitation of expert judgment was used to develop probability distributions for many accident progression, containment loading, structural response, and source term issues. The insights from the NUREG-1150 study have been used in several areas of reactor regulation, including the development of alternative radiological source terms for evaluating design-basis accidents at nuclear reactors.

4.2 The MELCOR Code

The MELCOR code, a fully integrated, engineering-level computer code, has, as its primary purpose, to model the progression of accidents in LWR NPPs, as well as in nonreactor systems (e.g., spent fuel pool, dry cask). Current uses of MELCOR include estimation of fission product source terms and their sensitivities and uncertainties in a variety of applications. MELCOR is a modular code comprising three general types of packages: (1) basic physical phenomena (i.e., hydrodynamics, heat and mass transfer to structures, gas combustion, aerosol and vapor physics), (2) reactor-specific phenomena (i.e., decay heat generation, core degradation, ex-vessel phenomena, sprays, and engineering safety systems), and (3) support functions (thermodynamics, equations of state, other material properties, data-handling utilities, and equation solvers). As a fully integrated code, MELCOR models all major systems of a reactor plant and their important coupled interactions.

Figure 4 shows the MELCOR code integration of models for important phenomena previously treated in separate effects codes.

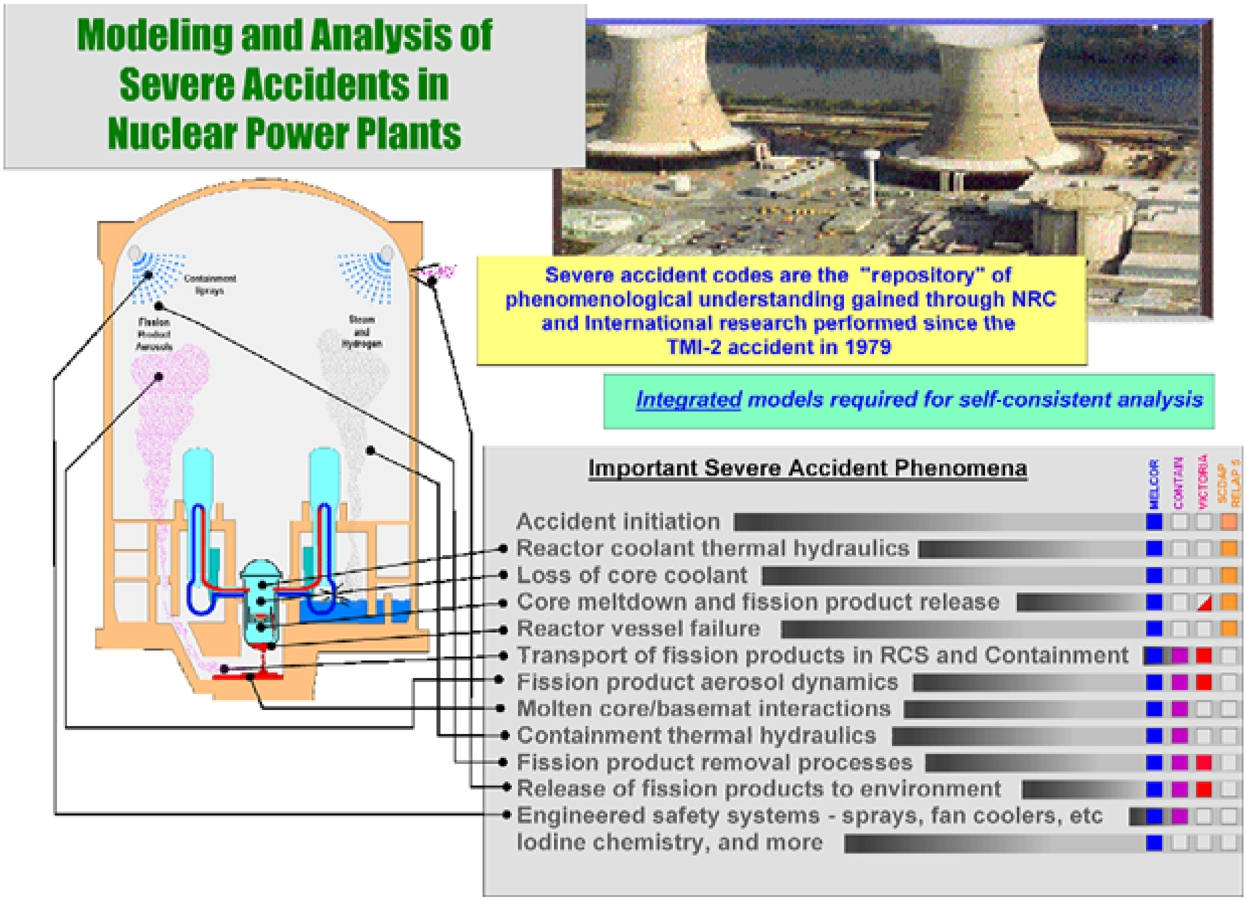

Figure 4 MELCOR integration of separate effects codes

The scope of MELCOR includes the following:

- thermal-hydraulic response of the RCS, reactor cavity, containment, and confinement buildings
- core uncovery (loss of coolant), fuel heatup, cladding oxidation, fuel degradation (loss of rod geometry), and core material melting and relocation
- heatup of reactor vessel lower head from relocated core materials and the thermal and mechanical loading and failure of the vessel lower head and transfer of core materials to the reactor vessel cavity
- core-concrete attack and ensuing aerosol generation
- in-vessel and ex-vessel hydrogen production, transport, and combustion
- fission product release (aerosol and vapor), transport, and deposition

- behavior of radioactive aerosols in the reactor containment building, including scrubbing in water pools and aerosol mechanics in the containment atmosphere, such as particle agglomeration and gravitational settling

- the impact of engineered safety features on thermal-hydraulic and radionuclide behavior

Most MELCOR models are mechanistic, and the use of parametric models is limited to areas of high phenomenological uncertainty where no consensus exists concerning an acceptable mechanistic approach. Current use of MELCOR often includes uncertainty analyses and sensitivity studies. To facilitate this, many of the mechanistic models have been coded with optional adjustable parameters. This does not affect the mechanistic nature of the modeling, but it does allow the analyst to easily address questions of how particular modeling parameters affect the course of a calculated transient. MELCOR does not use core radioactive nuclide inventories; rather, it uses masses and decay heats of chemical element groups. Appropriate code calculations for specific fuel and core design are carried out to the burnup of interest to provide the initial core inventories for MELCOR severe accident analysis (see Section 4.3.1).

After the completion of Version 1.8.1 in 1991, the NRC commissioned a peer review using recognized experts from national laboratories, universities, and the MELCOR user community [61]. The charter of the MELCOR peer review committee was to (1) provide an independent assessment of the MELCOR code through a peer review process, (2) determine the technical adequacy of the MELCOR code for the complex analyses it is expected to perform, and (3) issue a final report describing its technical findings. The committee offered a set of major findings that covered the various physics model numerics, missing models, modeling deficiencies, code assessment, and documentation. The NRC incorporated the findings into the research plan that governed the subsequent code development.

In 2000, the NRC began reducing the number of codes that it actively maintained by consolidating the CONTAIN, SCDAP/RELAP5, and VICTORIA code functionality and models into MELCOR. The assessment of MELCOR parity with CONTAIN showed that MELCOR results are comparable to CONTAIN. A comprehensive parity study of the MELCOR code with SCDAP/RELAP5 is ongoing. The assessment of fission product chemistry and transport is currently supported by foreign experiments (especially those from the Phebus facility in France). Hence, the scope of the evaluation of parity of the MELCOR to the VICTORIA code not only includes the phenomena treated in VICTORIA but also new experimental findings.

4.3 MELCOR Modeling Approach

Section 4.3.1 presents a high-level description of MELCOR models used for the SOARCA project. Existing MELCOR models for Surry and Peach Bottom were updated to current state-of-the-art modeling practices, as well as the latest version of the MELCOR code (Version 1.8.6). More detailed information describing the plant models is in the plant-specific analysis reports (i.e., NUREG/CR-7110, Volumes 1 and 2, for Peach Bottom and Surry, respectively).

The modeling and prediction of accident progression and radiological release in a severe accident requires the integration of a number of phenomenological models to address a range of thermal-hydraulic, materials, structural, and fission product behavior, as well as models for component (e.g., safety relief valve) behavior. Section 4.3.2 describes the procedure to define the best practices approach to modeling important and uncertain phenomena. NUREG/CR-7008, "Best Practices for Simulation of Severe Accident Progression at Nuclear Power Plants" [6], provides a more detailed description of the best practices modeling approach. At the beginning of the SOARCA project, an independent review of MELCOR best practices modeling provided greater assurance of the technical soundness of the analytical modeling [42]. The NRC used that review to identify and incorporate subsequent modeling insights and improvements before the start of plant analyses. Moreover, members of the SOARCA peer review committee recommended additional sensitivity analyses to explore specific modeling issues that were viewed as both uncertain and potentially important to risk. These analyses, discussed in detail in NUREG/CR-7110, Volumes 1 and 2, help confirm that the modeling of MELCOR best practices is sound.

Section 4.3.3 summarizes some recent changes to the modeling of radionuclide release and cesium speciation, which is important to the source term results. Finally, Section 4.3.4 describes the methodology for calculating the radionuclide inventory.

4.3.1 Plant Models

The SOARCA program updated the MELCOR models for Peach Bottom and Surry to the most recent version of the MELCOR code.[4] The scope of the models included the following:

- detailed five-ring reactor vessel models
- representation of the RCS (and secondary system through the main steam isolation valve for Surry)
- representation of the primary containment
- representation of the Peach Bottom reactor building and the Surry Safeguards and Auxiliary Buildings, and ventilation and filter systems, which were radionuclide pathways in the ISLOCA scenario
- representation of the emergency core cooling systems (and the auxiliary feedwater system for Surry)
- representations of the emergency portable water-injection systems

[4] All SOARCA calculations used MELCOR Version 1.8.6.

The best practices updates to each input deck specified the following new models for both plants for these important but uncertain phenomena or equipment responses:

- Safety relief valve failure modeling addressing stochastic and high-temperature failure modes.
- An additional thermomechanical fuel collapse model for heavily oxidized fuel following molten Zircaloy breakout.
- Enhanced lower plenum coolant debris heat transfer that recognizes breakup and multidimensional cooling effects not present in the one-dimensional countercurrent flooding model in older versions of MELCOR (e.g., [23]).
- Updated, plant-specific chemical element masses and decay heats (see Section 4.3.4).
- A new Oak Ridge National Laboratory (ORNL) Booth chemical element release model and new cesium speciation model (see Section 4.3.3).
- A new turbulent deposition model for aerosol deposition in piping systems. NUREG/CR-7110, Volume 2 discusses this new model and its validation.
- Vessel failure based on gross failure [24] using the improved one-dimensional creep rupture model with the new hemispherical head model and radial heat transfer between lower head conduction node segments. A more complete discussion of this model is presented in NUREG/CR-7008 [6] and the MELCOR manual [31]. A penetration failure model was not used because the timing differences between gross lower head failure and penetration failure with the available penetration model are not significant to the overall accident progression (i.e., minutes difference). Also, Sandia lower head failure tests showed that gross creep rupture of the lower head was measured to be the most likely mechanism for vessel failure [24].
- Enhanced ex-vessel core debris heat transfer that reflects multidimensional effects and rates measured in MACE tests [25].

Sections 4.3.1.1 and 4.3.1.2 summarize the SOARCA program's recent enhancements to the MELCOR Peach Bottom and Surry models, respectively.

4.3.1.1 Peach Bottom MELCOR Model

Brookhaven National Laboratory originally developed the Peach Bottom MELCOR plant model for code Version 1.8.0. J. Carbajo at ORNL subsequently adopted the model to study differences in fission product source term behavior predicted by MELCOR 1.8.1 and those generated for use in NUREG-1150 [2] using the STCP [26]. Starting in 2001, SNL has made considerable refinements to the BWR/4 core nodalization to support the developmental assessment and release of MELCOR 1.8.5. These refinements concentrated on the spatial nodalization of the reactor core (both in terms of fuel and structural material and hydrodynamic volumes) used to calculate in-vessel melt progression.

Subsequent work in support of several NRC research programs has motivated further refinement and expansion of the BWR/4 model in four broad areas. The first area involved the addition of models to represent a wide spectrum of plant design features, such as safety systems, to broaden the capabilities of MELCOR simulations to apply to a wider range of severe accident sequences. These enhancements include the following:

- modifications of modeling features needed to achieve steady-state reactor conditions (recirculation loops, jet pumps, steam separators, steam dryers, feedwater flow, control rod drive hydraulic system, main steamlines, turbine/hotwell, core power profile)

- new models and control logic to represent coolant injection systems (RCIC, HPCI, residual heat removal, low-pressure core spray) and supporting water resources (e.g., CST with switchover)

- new models to simulate reactor vessel pressure management (safety relief valves, safety valves, automatic depressurization system, and logic for manual actions to effect a controlled depressurization if torus water temperatures exceed the heat capacity temperature limit)

The second area focused on the spatial representation of the containment and the reactor building. The drywell portion of containment has been subdivided to distinguish thermodynamic conditions internal to the pedestal from those within the drywell itself. Also, refinements have been added to the spatial representation and flow paths within the reactor building. A containment failure model is included that accounts for leakage around the drywell head flange, leakage caused by elevated drywell temperature, and leakage caused by drywell melt-through (see NUREG/CR-7110, Volume 1, Section 4.6). The third area focused on bringing the model up to current "best practice" standards for MELCOR 1.8.6 (see Section 4.3.2). The fourth area of model improvements included a new radionuclide inventory and decay heat based on the recent plant operating history (see Section 4.3.4).

Although not new for SOARCA, the MELCOR Peach Bottom model includes a multiregion ex-vessel debris spreading model. The debris spreads according to its temperature relative to the solidus and liquidus temperatures of the concrete and the debris height. If the debris spreads against the drywell liner steel wall, and if the debris temperature is above the carbon steel melting temperature, the liner will fail.

The potential for creep rupture of a BWR main steamline (i.e., piping or RPV nozzle) was added to the Peach Bottom model developed for SOARCA.

NUREG/CR-7110, Volume 1, more fully describes the MELCOR Peach Bottom model. Figure 5 shows the MELCOR nodalization diagrams for Peach Bottom.

4.3.1.2 Surry MELCOR Model

In 1988, Idaho National Engineering Laboratory originally generated the Surry MELCOR model applied in this study. SNL periodically updated it (1990 to present) to test new models, advancing the state of the art in modeling PWR accident progression and providing support to the NRC for analyses of various issues that could affect operational safety. Significant changes were made during the last several decades in the approach to modeling core behavior and core melt progression, as well as the nodalization and treatment of coolant flow within the RCS and reactor vessel. In 2002, the reactor vessel and RCS nodalization were updated using the SCDAP/RELAP5 Surry model to include a five-ring vessel nodalization and countercurrent hot-leg representation for natural circulation flow [27]. The current MELCOR Surry model is a culmination of these efforts.

In preparation for the SOARCA analyses described in this report, the model was further refined and expanded in three areas. The first area is an upgrade to core modeling in MELCOR Version 1.8.6. These enhancements include the following:

- a hemispherical lower head model that replaces the flat-bottom cylindrical lower head model
- new models for the core former and shroud structures that are fully integrated into the material degradation modeling, including separate modeling of debris in the bypass region between the core barrel and the core shroud
- models for simulating the formation of molten pools in both the core and lower plenum, crust formation, convection in molten pools, stratification of molten pools into metallic and oxide layers, and partitioning of radionuclides between stratified molten pools
- a reflood quench model that separately tracks the component quench front and the quenched and unquenched temperatures
- a control rod aerosol release model
- addition of the new ORNL Booth radionuclide release model for modern high-burnup fuel

The second area focused on the addition of user-specified models to represent a wide spectrum of plant design features and safety systems to broaden the ability of MELCOR to apply to a wider range of severe accident sequences. These enhancements included the following:

- update of the containment leakage model to include nominal leakage and leakage caused by containment overpressure (see NUREG/CR-7110, Volume 2, Section 4.8)
- update of core degradation modeling practices

- models of the individual primary system and secondary system relief valves with failure logic for rated and degraded conditions

- update of the containment flooding characteristics

- heat loss from the reactor to the containment

- separate motor driven auxiliary feed water and TDAFW models with control logic for plant automatic and operator cooldown responses

- new TDAFW models for steam flow, flooding failure, and performance degradation at low pressure

- nitrogen discharge model for accumulators

- update of the fission product inventory, the axial and radial peaking factors, and an extensive fission product tracking control system

- improvements to the modeling of natural circulation in the hot leg and steam generator and the potential for creep rupture

NUREG/CR-7110, Volume 2, more fully describes the MELCOR Surry model. Figure 5 shows the MELCOR nodalization diagrams for Surry.

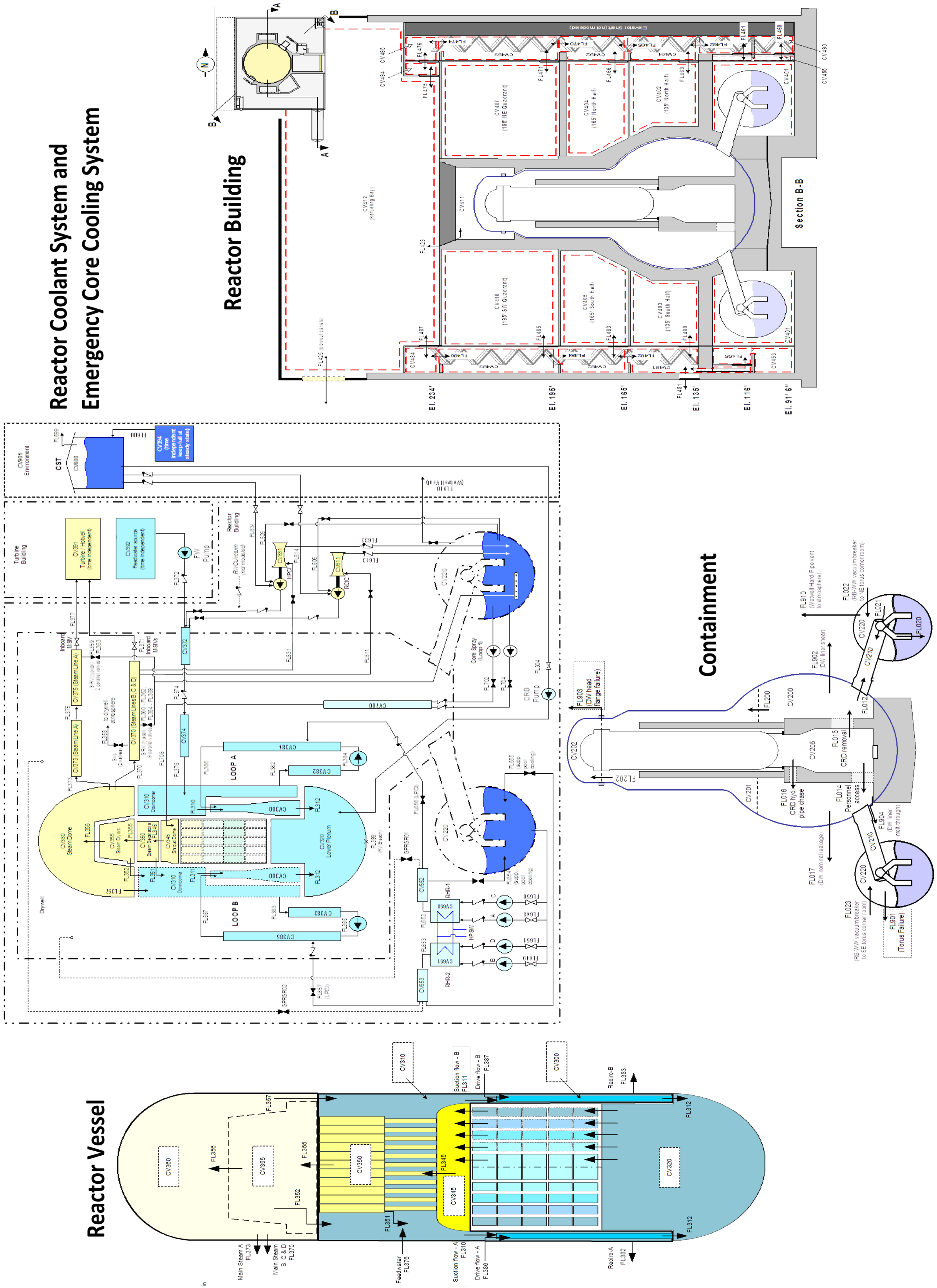

Figure 5 The Peach Bottom MELCOR nodalization

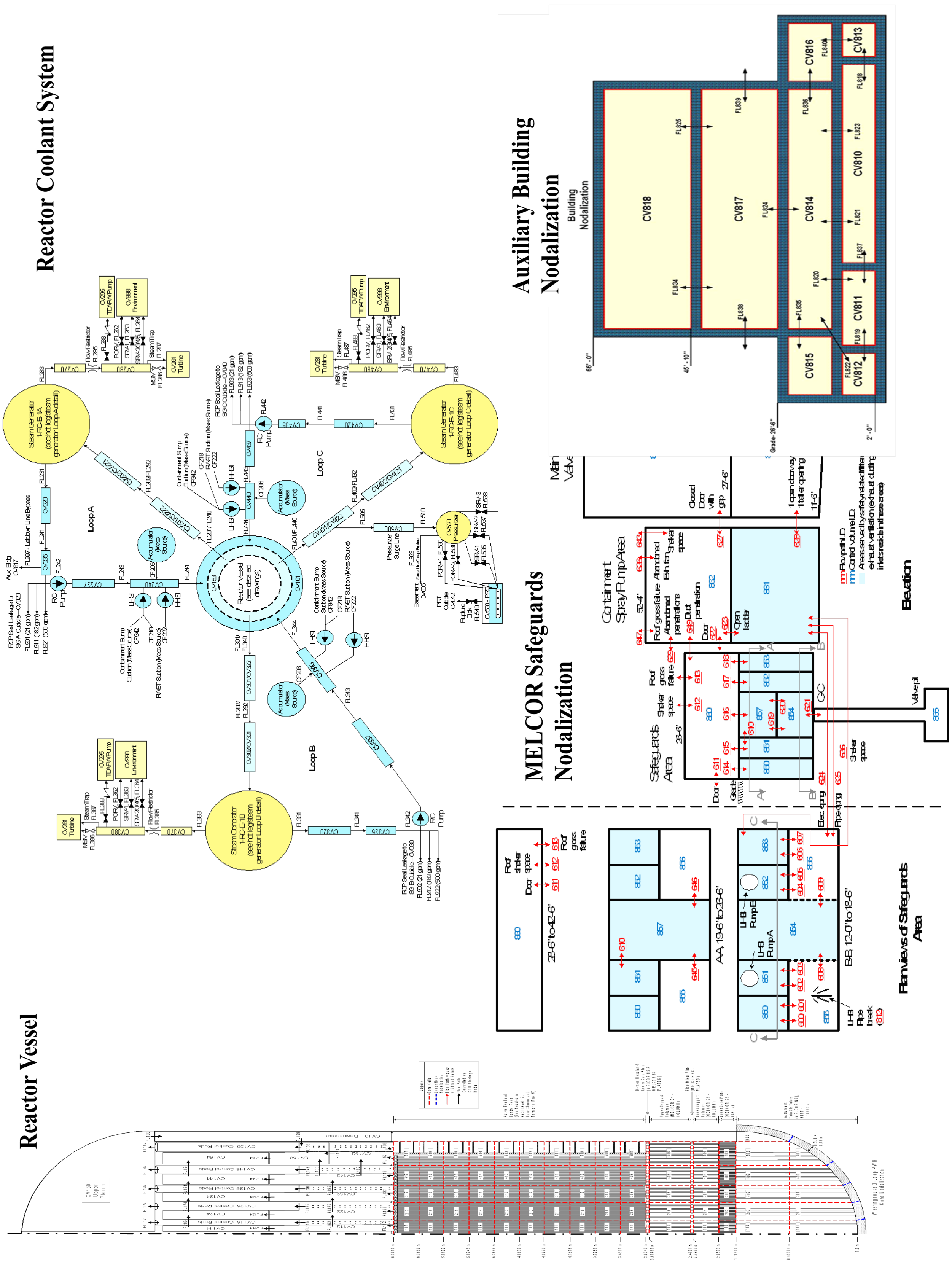

Figure 6 The Surry MELCOR nodalization

4.3.2 Best Modeling Practices

The SOARCA project's integrated modeling of the accident progression and offsite consequences uses both state-of-the-art computational analysis tools and best modeling practices drawn from the collective body of knowledge on severe accident behavior generated over the past several decades of research.

The MELCOR 1.8.6 computer code embodies much of this knowledge and was used for the accident and source term analysis. MELCOR includes capabilities to model the two-phase thermal-hydraulics, core degradation, fission product release, transport, deposition, and containment response. The SOARCA analyses include operator actions and equipment performance issues as prescribed by the sequence definition and mitigative actions. The MELCOR models are constructed using plant data, and the operator actions were developed based on tabletop exercises during site visits. The code models and user-specified modeling practices represent the current best practices.

While much has been learned through extensive research, uncertainties exist in understanding phenomena associated with severe accident progression and radionuclide transport. Consistent with the stated objective of SOARCA, phenomena were modeled using realistic characterization of phenomena and events. The accident progression analysts developed a list of key uncertain phenomena that can have a significant effect on the progression of the accident. Plant-specific reports for Peach Bottom (NUREG/CR-7110, Volume 1) and Surry (NUREG/CR-7110, Volume 2) outlined each issue and identified a modeling approach or base case values. NUREG/CR-7008 [6] discusses the specific modeling practices.

The SOARCA project excluded several early containment failure modes of historical interest because of their assessed low likelihood of occurrence. These include the following:

- Alpha mode containment failure would be caused by an in-vessel steam explosion during melt relocation that simultaneously fails the vessel and the containment. A group of experts in this field, referred to as the Steam Explosion Review Group, concluded, in a position paper published by the Nuclear Energy Agency Committee on the Safety of Nuclear Installations [28], that the alpha-mode failure issue for Western-style reactor containment buildings can be considered resolved from a risk perspective, having little or no significance to the overall risk from an NPP.

- Direct containment heating would be caused by containment failure in PWR containments. NRC research has shown that an early failure of the PWR RCS caused by high-temperature natural circulation will likely depressurize the RCS before vessel failure. Importantly, extensive NRC testing and analyses have also shown that, in the unlikely event of a high-pressure vessel failure, early containment failure caused by direct containment heating is very unlikely, with some variation depending on plant design [29]. In the case of Surry, the research concluded that no feasible likelihood exists of failing the containment.

- Early containment failure would be caused by drywell liner melt-through in a wet cavity in Mark I containments (e.g., Peach Bottom). Through a detailed assessment of the issue, the research concluded that, in the presence of water, the probability of early containment failure by melt-attack of the liner is so low as to be considered physically unreasonable [30].

At the start of the SOARCA project, a panel of experts reviewed the proposed modeling approach for SOARCA analyses during a public meeting sponsored by the NRC on August 21–22, 2006, in Albuquerque, NM. The panel examined the best modeling practices for the application of MELCOR to realistically evaluate accident progression and source term. The panel also reviewed a set of code enhancements and considered the SOARCA project in general.

4.3.3 Radionuclide Modeling

The radionuclide modeling was updated in the Peach Bottom and Surry models to apply a more mechanistic radionuclide release model (i.e., the ORNL-Booth model [31]) based on assessments of recent radionuclide release tests. These assessments identified an alternative set of Booth diffusion parameters recommended by ORNL (ORNL-Booth) [32] that produced significantly improved release signatures for cesium and other fission product groups. Some adjustments to the scaling factors in the ORNL-Booth model were made for selected fission product groups including uranium dioxide (UO_2), molybdenum (Mo), and ruthenium (Ru) to gain better comparisons with FPT-1 data [33]. The adjusted model, referred to as "Modified ORNL-Booth," was subsequently compared to original ORNL VI fission product release experiments and to more recently performed VERCORS tests [34], and the comparisons were as favorable or better than the original CORSOR-M MELCOR default release model. These modified ORNL-Booth parameters were introduced into the MELCOR code as new defaults for the SOARCA project.

Although the analysis of the FPT-1 test with the ORNL-Booth parameters obtained significant improvements in release behavior, some additional modification to the MELCOR release model was pursued. Evidence from the Phebus experiments increasingly indicates that the dominant chemical form of released Cs is that of Cs_2MoO_4. This is based on deposition patterns in the Phebus experiment, where Cs is judged to be in aerosol form at 700 degrees Celsius, which explains deposits in the hot upper plenum of the Phebus test section and deposition patterns in the cooler steam generator tubes. In recognition of response, a Cs_2MoO_4 radionuclide class was defined with the vapor pressure Cs_2MoO_4 and the release coefficients developed for Cs. The Mo vapor pressure is so exceedingly low that the net release is limited by the vapor pressure transport term. Because there is significantly more Mo than Cs in the radionuclide inventory, only a portion of the Mo was added to the new Cs_2MoO_4 radionuclide class.

There are 69 isotopes in the treatment of consequences considered in the MACCS2 analysis, as described in NUREG/CR-7110, Volume 1. These isotopes are grouped into a set of nine chemical classes in the MELCOR analyses that generated the source terms used in the SOARCA analyses. Since release fractions are calculated by MELCOR at the level of chemical classes, it is reasonable and useful to examine how these same chemical classes influence the evaluation of risk. Volumes 1 and 2 of NUREG/CR-7110 discuss the importance of chemical classes.

The radionuclide input was reconfigured to (1) represent the dominant form of Cs as Cs_2MoO_4, (2) represent the dominant form of iodine (I) as CsI, and (3) represent the gap inventories consistent with the NUREG-1465 recommendations [14]. The MELCOR radionuclide transport, deposition, condensation and evaporation, and scrubbing models were all activated. The model for chemisorption of Cs to stainless steel was activated. In addition, the hygroscopic coupling of the steam or fog condensation or evaporation thermal-hydraulic solutions to the airborne aerosol size and mass was also activated [31].

4.3.4 Radionuclide Inventory

One important input to MELCOR is the initial mass of the radionuclides in the fuel and their associated decay heat [31]. These values are important to the timing of initial core damage and the location and concentration of the radionuclides in the fuel. The radioisotopes in a nuclear reactor come from three primary sources: (1) fission products, which are the result of fissions in either fissile or fissionable material in the reactor core, (2) actinides, which are the product of neutron capture in the initial heavy metal isotopes in the fuel, and (3) radioactive decay of these fission products and actinides. Integrated computer models such as the TRITON sequence in SCALE exist to capture all of these interrelated physical processes, but they are intended primarily as reactor physics tools [35]. As such, their standard output does not provide the type of information needed for SOARCA. Therefore, this report describes a method for deriving the needed information. It is important to note that no changes to the physics codes were needed. The method described here merely extracts additional output from the TRITON sequence and combines it in a way that makes it useful for the SOARCA project.

4.3.4.1 Methods

Reactor physics codes implicitly account for both of the physical parameters of interest for SOARCA (i.e., decay heat power and radionuclide inventories), but they do not provide a mechanism to easily extract and combine these results. This section will describe the tools used to calculate the radioisotopic inventory and a new code developed to properly combine these results for use in the SOARCA calculations. The results were combined in a manner so as to capture actual plant operating data.

The TRITON sequence from SCALE 5.1 was used to develop input data for MELCOR. TRITON allows detailed two-dimensional calculations of reactor fuel, including the ability to deplete fuel to a user-defined level of accuracy. TRITON accurately models curvilinear surfaces such as cylindrical fuel rods and allows the fuel to be burned down to the subpin-cell level. There is no requirement to perform any homogenization of the two-dimensional geometry. TRITON allows for accurate depletion of highly self-shielded fuel such as poison pins. For more information, refer to the SCALE documentation [36].

The BLEND3 code was developed from previous work performed by ORNL, and its capabilities were extended for this study. BLEND3 uses the reactor-specific fuel loading from three different cycles, the nodal exposure, and the assembly-specific power data from the licensee to derive node-averaged radioisotopic inventories. TRITON uses generic fuel assembly data and ties them to specific reactor operating conditions. Then, BLEND3 performs the following tasks.

First, for a given node, BLEND3 identifies which specific power ORIGEN output files are assigned to the specified input power. Second, for three different cycles of fuel, BLEND3 interpolates a radioisotopic inventory from the relevant ORIGEN output files. Finally, using the input volume fractions for the three different cycles of fuel, BLEND3 creates a new, volumetrically averaged ORIGEN output file for the node for the specified input conditions.

The PRISM module from SCALE 5.1 was then used to drive ORIGEN decay calculations using the newly created averaged ORIGEN output files as input. PRISM is a SCALE utility module that allows the user to automate the execution of a series of SCALE calculations.

4.3.4.2 Peach Bottom Model

The Peach Bottom model is based on the Global Nuclear Fuel (GNF) 10×10 (GE-14C) fuel assembly. The GNF 10×10 is representative of a limiting fuel type actually being used in commercial BWRs. Figure illustrates the GNF 10×10 model. The axial nodalization of the core is designed, in part, to account for changes in material composition and mass along the axial length of a typical fuel assembly. For example, some BWR fuel assembly designs (modern 10×10 assemblies, for example) incorporate fuel rods of different lengths within a single assembly. As a result, the amount of UO_2 and other constituents can differ at the top and bottom of an assembly. Discrete locations of fuel rod spacers along the axial height of an assembly also affect local Zircaloy mass. The distribution of material mass within the axial nodalization of the core takes these variations into account.

At nine different specific power histories, 27 different TRITON runs were performed to model three different cycles of fuel. The specific power histories ranged from 2 megawatt-days per metric ton of uranium (MWd/MTU) to 45 MWd/MTU to cover all expected BWR operational conditions. For times before the cycle of interest, an average specific power of 25.5 MWd/MTU was used. For example, for second cycle fuel, the fuel was burned for its first cycle using 25.5 MWd/MTU, allowed to decay for an assumed 30-day refueling outage, and then nine different TRITON calculations were performed with specific powers ranging from 2 to 45 MWd/MTU). The BLEND3 code was then applied to each of the 50 nodes in the MELCOR model using the average specific powers and volume fractions. Once new libraries for each of the 50 nodes in the model were generated, the final step in the procedure was to deplete each node for 48 hours. The decay heats, masses, and specific activities as a function of time were processed and applied as input data to MELCOR to define decay heat and the radionuclide inventory. The SOARCA application, in keeping with the intent of using best estimate approaches, based the Peach Bottom fuel analysis of decay power and radionuclide inventories on the assumption that the accident occurs at a point midway in a recent fuel cycle.

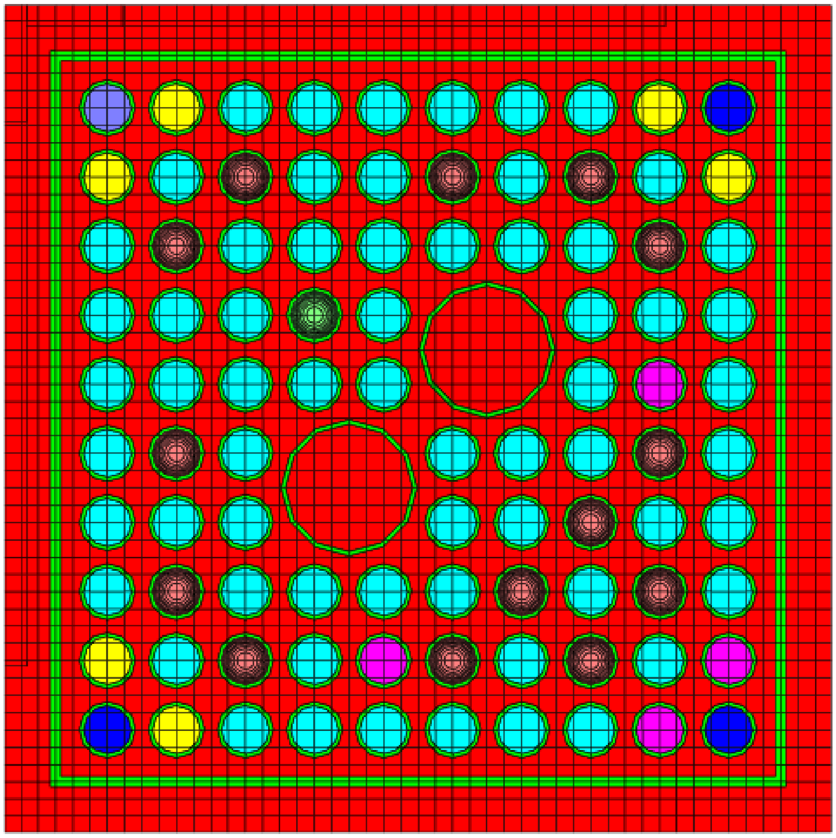

Figure 7 Schematic of modeling detail for BWR GNF 10×10 assembly

4.3.4.3   Surry Model

Previously, detailed input was developed for Surry in a separate NRC program investigating the source term from high-burnup uranium fuel. This study used the same methodology as the Peach Bottom model (Section 4.3.4.2) but extended the burn-up of the lead assembly to the licensing limit (i.e., above current best-estimate practices). Based on comparisons to the Peach Bottom decay heat, the best-estimate, midcycle decay power for a recent Surry fuel cycle is expected to be about 17–18 percent lower than that used in the SOARCA MELCOR analyses for Surry.

4.3.4.4   Evaluation of the Results

Very few measurements of decay heat exist, and those that do are not directly relevant to this study. Therefore, the discussion of the decay heat predictions will be limited to a comparison to previously published work. RG 3.54, “Spent Fuel Heat Generation in an Independent Spent Fuel Installation” [37] summarizes a source of decay heat predictions, and results from RG 3.54 will be used to assess the predictions in the current study. Decay heat for two decay times will be used as a check on the consistency of the results presented in this study. By interpolation of tables in RG 3.54 for a specific power of 27 MW/MTU, decay powers at 1 and 2 years following shutdown of 9.3 W/kgU and 5.1 W/kgU, respectively, are calculated. Using the results from the Peach Bottom calculations, the corresponding decay powers are 8.92 W/kgU and 4.734 W/kgU. The maximum difference between results is about 8 percent, which is considered acceptable given the best estimate nature of the SOARCA study compared to the methods used to generate the tables in RG 3.54.

A quantitative discussion of the radioisotopic predictions presented in this study would be of limited use, given the cycle-specific nature of this work. However, it is beneficial to discuss the relevant SCALE assessment. Specifically, the TRITON module has been assessed by M.D. DeHart and S.M. Bowman [38], S.M. Bowman and D.F. Gill [39], and I. Germina and I.C. Gauld [40]. These assessment reports use data from Calvert Cliffs, Obrigheim, San Onofre, and Trino Vercelles PWRs. The third report [40] summarized comparisons to decay heat measurements from four different BWR assemblies.

5.0 OFFSITE CONSEQUENCE ANALYSES

MACCS2 [41] is a consequence analysis code for evaluating the impacts of atmospheric releases of radioactive aerosols and vapors on human health and the environment. It includes all of the relevant dose pathways: cloudshine, inhalation, groundshine, and ingestion. Because it is primarily a PRA tool, it accounts for the uncertainty in weather that is inherent to an accident that could occur at any point in the future. WinMACCS is a user-friendly front end to MACCS2 that facilitates selection of input parameters and sampling of uncertain input parameters, and it performs postprocessing of results. The final SOARCA calculations use WinMACCS Version 3.6. MACCS2 is still the computational engine underlying WinMACCS.

The SOARCA offsite consequence predictions used MACCS2 Version 2.5. This version includes a number of improvements to the original MACCS2 code, which can be categorized as follows:

- atmospheric transport and dispersion modeling improvements (e.g., morning and afternoon mixing heights, alternative Briggs plume rise model, and alternative long-range plume spreading model)
- capability to describe wind directions in 64 compass directions (instead of 16)
- increased limits on several input parameters (e.g., a limit of 200 plume segments instead of the previous limit of 4)
- up to 20 emergency-phase cohorts (instead of the original limit of 3) to describe variations in emergency response by segments of the population
- enhancements in the treatment of evacuation speed and direction to better reflect the spatial and temporal response of individual cohorts
- capability to run on a cluster of computers instead of an individual processor
- addition of several new options for LCF dose response (i.e., user-input yearly truncation value, user-input yearly truncation value with a lifetime restriction, and a piece-wise linear model)

An expert panel reviewed the MACCS2 code and modeling choices in August 2006, before specific work on Surry and Peach Bottom began. This expert panel review and the NRC staff recommendations influenced much of the development undertaken specifically to support the SOARCA work [42].

Subsequent parts of this chapter describe specific aspects of the consequence modeling in SOARCA that depart from previous studies such as NUREG-1150 [2].

5.1 Weather Sampling

The weather sampling strategy adopted for SOARCA uses the nonuniform weather-binning approach in MACCS2. This approach, which allows the user to specify a different number of random samples to be chosen from each bin, has been available since MACCS2 was first released [41] but was not commonly used in the past. Weather binning is an approach used in MACCS2 to categorize similar sets of weather data based on windspeed, stability class, and the occurrence of precipitation. The SOARCA project chose this sampling strategy to improve the statistical representation of the weather, as is further discussed below.

The standard way of defining weather bins originated in the NUREG-1150 [2] analyses. A set of 16 weather bins differentiates stability classes and wind speeds. An additional 20 weather bins include all weather trials in which rain occurs before the initial plume segment travels a distance of 32 kilometers (20 miles). The bins differentiate rain intensity and the distance the plume travels before rain begins. The parameters used to define the rain bins are the same as those used in NUREG-1150 and documented in the MACCS2 User's Manual [41]. Because the strategy provides for weighting the particular trials chosen (based on the number of samples in the bin and the number of samples requested), the particular choice of a binning strategy is not important (provided a sufficient number of samples is chosen). However, a well-chosen binning strategy will reduce the number of samples required for adequate statistical precision. The binning strategy used in NUREG-1150 and for SOARCA ensures that the rain cases, which are only a fraction of the full year's data, are adequately sampled, with the weighting factors used in the code accounting for the prevalence in the weather record.

For the nonuniform weather sampling strategy approach for SOARCA, the number of trials selected from each bin is the maximum of 12 trials and 10 percent of the number of trials in the bin. Some bins contain fewer than 12 trials. In those cases, all of the trials within the bin are used for sampling. This strategy results in roughly 1,000 weather trials for both Peach Bottom and Surry.

Previous calculations, such as NUREG-1150, used about 125 weather trials, including an additional strategy—rotation—to account for the probability that the wind might have been blowing in a different direction when the release began. This strategy uses wind-rose data constructed from the annual weather file to determine the probability that the wind might have been in any of the compass directions. The strategy used at the time of NUREG-1150 leveraged the weather data to get 125×16=1,750 results for the computational price of 125, but at a cost that the individual results are not truly independent. For the strategy chosen here, the trials are independent.

MACCS2 does not allow the use of rotation in concert with the network evacuation option; therefore, rotation was not an option for SOARCA. The strategy adopted for SOARCA was a compromise between obtaining adequate statistical significance and keeping central processing unit time at a reasonable level.

5.2 Weather Data

The SOARCA project used 1 year of hourly meteorological data for each site (8,760 data points per site for each meteorological parameter). This was primarily accomplished through a cooperative effort, with the licensee using onsite meteorological tower observations. Each licensee provided 2 years of weather data. The licensees measured and reported hourly precipitation directly. Temperature measurements at two elevations on the site meteorological towers provided stability class data. The project based the specific year of data chosen for each reactor on data recovery (greater than 99 percent being desirable) and proximity to the target year for SOARCA, which is 2005. Different trends (e.g., wind-rose pattern and hours of precipitation) between the years were estimated to have a relatively minor (less than 25 percent) effect on the final results. The next subsection discusses the specific details of the weather data.

For the weather record years and the particular data used in SOARCA, the recovery of data was in excess of 90 percent. The missing data were bridged over using the hourly records before and after by employing "Procedures for Substituting Values for Missing NWS [National Weather Service] Meteorological Data for Use in Regulatory Air Quality Models," dated July 7, 1992 [43]. The meteorological data parameters were formatted for the MACCS2 computer code.

The NRC staff used the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data," issued July 1982 [44] to perform quality assurance evaluations of all meteorological data presented. Further review used computer spreadsheets. The NRC staff ensured a joint data recovery rate in the 90th percentile, which is in accordance with RG 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants," Revision 1, issued March 2007 [45] for the wind speed, wind direction, and atmospheric stability parameters. In addition, it evaluated atmospheric stability to determine if the time of occurrence and duration of reported stability conditions were generally consistent with expected meteorological conditions (e.g., neutral and slightly stable conditions predominated during the year with stable and neutral conditions occurring at night and unstable and neutral conditions occurring during the day). The mixing height data came from the U.S. Environmental Protection Agency's (EPA's) SCRAM database[5] (using data from the years 1984–1992). Data needed for MACCS2 includes 10-meter wind speed, 10-meter wind direction in 64 compass directions, stability class (using the Pasquill-Gifford scale and representative values of 1–6 for stability classes A–F/G (see Section 5.2.1)), hourly precipitation, and diurnal (morning and afternoon) seasonal mixing heights.

All of the SOARCA consequence analyses included boundary weather, but it was imposed beyond the outer boundary (50 miles or ~80.5 kilometers) for which results are reported. Thus, the choice of boundary weather had no influence on the consequence results that are reported. Appendices in the companion Peach Bottom and Surry reports contain the specific parameters chosen to describe the boundary weather.

5 The EPA SCRAM Web site is http://www.epa.gov/scram001/mixingheightdata.htm.

5.2.1 Summary of Weather Data

Table 2 presents a summary of the meteorological statistical data and shows that the annual average ground-level wind speeds were generally low, ranging from 2.02 to 2.27 meters per second (m/s) at Surry and 2.12 to 2.17 m/s at Peach Bottom. The atmospheric stability frequencies were consistent with expected meteorological conditions. The neutral and slightly stable conditions predominated during the year, with stable and neutral conditions occurring at night and unstable and neutral conditions occurring during the day.

Figure 8 and Figure 9 show the wind direction (direction the wind blows toward) and atmospheric stability (unstable,[6] neutral,[7] and stable[8]) data for the years that were actually used in the consequence analyses (i.e., 2006 for Peach Bottom and 2004 for Surry). The MACCS2 calculations used the Pasquill-Gifford stability classes. These classes were only parsed into unstable (A–C), neutral (D), and stable (E–F) conditions for Figure 8 and Figure 9 for comparisons with expected weather patterns.

Table 2 Statistical Summary of Raw Meteorological Data for SOARCA Nuclear Sites

Parameter		Peach Bottom Year 2005	Peach Bottom Year 2006	Surry Year 2001	Surry Year 2004†
Average Wind Speed (m/s)		2.17	2.12	2.02	2.27
Yearly Precipitation (hr)		588 (6.7%)	593 (6.8%)	388 (4.4%)	521 (5.9%)
Atmospheric Stability (%)	Unstable	21.43	20.56	7.09	3.94
	Neutral	63.97	62.34	69.67	77.59
	Stable	14.60	17.10	23.24	18.47
Joint Data Recovery (%)		97.53	99.25	99.58	99.24

† Year 2004, used in the Surry meteorological analysis, is a leap year (8,784 total hourly data points versus 8,760 hourly data points for a regular annual period).

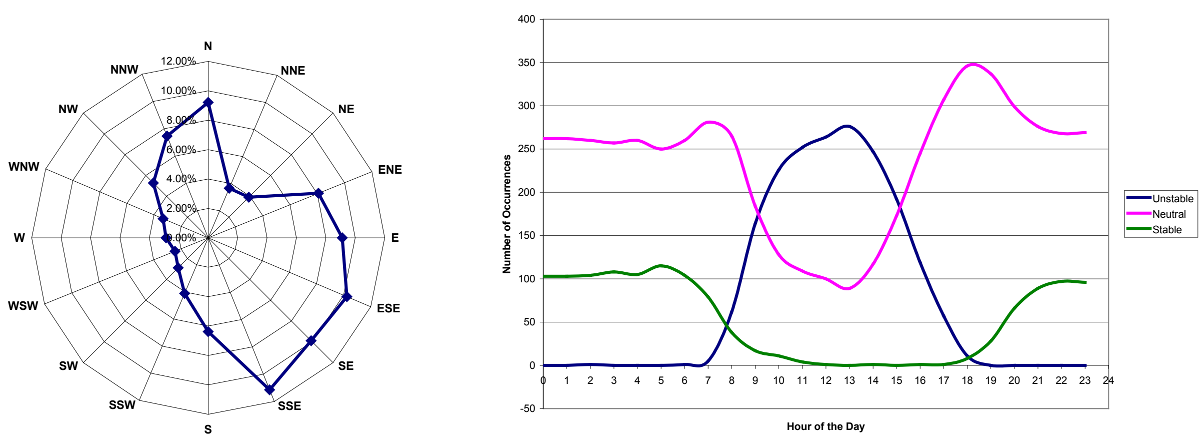

Figure 8 Peach Bottom—Year 2006—wind-rose and atmospheric stability chart

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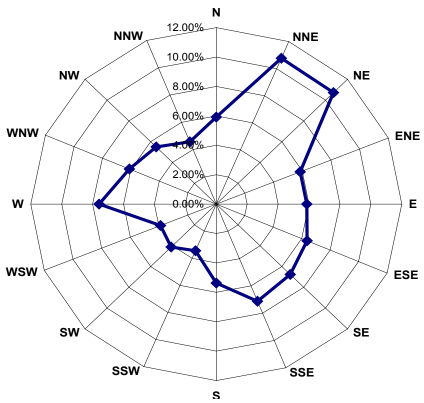
6 This corresponds to Pasquill-Gifford stability classes A, B, and C.
7 This corresponds to Pasquill-Gifford stability class D.
8 This corresponds to Pasquill-Gifford stability classes E, F, and G.



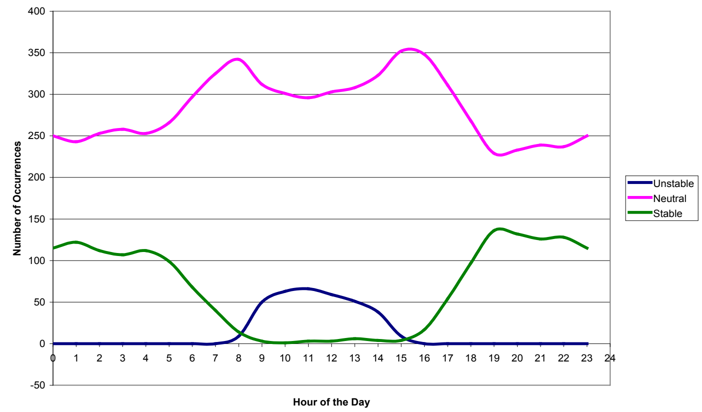


Figure 9 Surry—Year 2004—wind-rose and atmospheric stability chart

5.3 <u>Emergency Response Modeling</u>

An objective of the SOARCA project was to model emergency response in a more detailed and realistic manner using site-specific emergency planning information. The analysis included modeling of the timing of onsite and offsite decisions and implementation of protective actions applied to multiple population segments (called cohorts). Advances in consequence modeling—specifically the development of WinMACCS—made it easier to integrate protective action decision timing and response of the public into the consequence analysis, resulting in an evolutionary advancement over previous studies.

Emergency response programs for NPPs are designed to protect public health and safety in the unlikely event of a radiological accident. These emergency response programs are developed, tested, and evaluated and are in place as an element of the NRC's defense in depth policy. Detailed plans for onsite and offsite response are approved by the NRC and the Federal Emergency Management Agency respectively.

Offsite response organization emergency plans are required to include detailed evacuation plans for the 10-mile emergency planning zone (EPZ) [46]. Site-specific information was obtained from ORO emergency response plans to support development of timelines for protective action implementation. Site specific planning elements were modeled, for example whether evacuation of schools follows declaration of a site area emergency or a general emergency. The SOARCA project integrated response plan elements and a best estimate of protective action decision timing that was based upon actual biennial exercise history. Specific population cohorts were identified and their evacuation timing modeled. This detailed modeling was undertaken for the SOARCA project to improve the overall fidelity of the consequence analyses.

Figure shows the 10- and 20-mile radial distances around the Peach Bottom site.

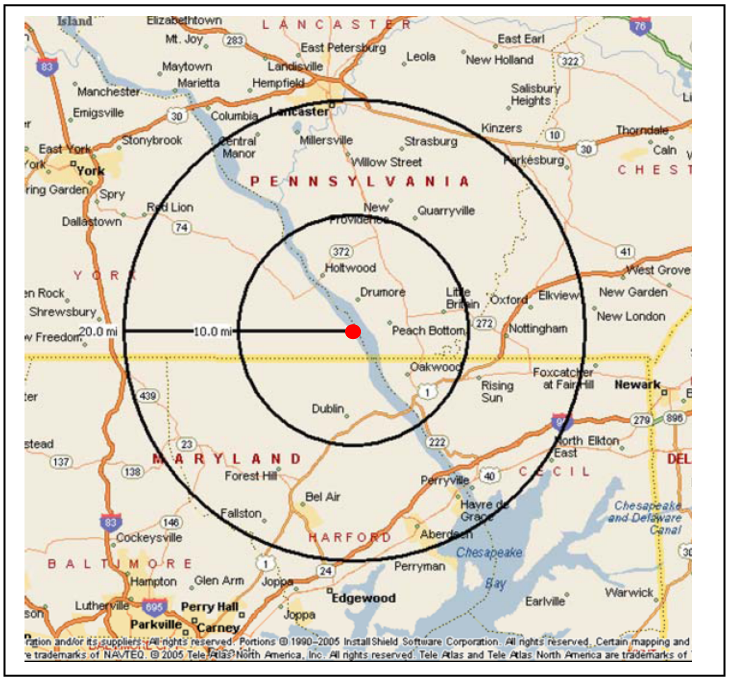


Figure 10 10- and 20-mile radial distances around the Peach Bottom site

The SOARCA project assessed the ORO protective action decision-making process as detailed in emergency plans and developed a best estimate of implementation of those decisions by ORO populations within the 10-mile EPZs. The project also assessed possible variations of emergency response for the two sites studied, including evacuation and sheltering of population groups beyond the EPZ to a distance of 20 miles from the plants. As discussed in NUREG-0654/FEMA-REP-1, Revision 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," issued November 1980 [46], detailed planning within 10 miles would provide a substantial base for expansion of response efforts, should this prove necessary. Any response beyond the EPZ was expected to be limited to areas where dose projections indicate protective actions are necessary. State or county response agencies would inform the OROs of these projections in areas beyond the EPZ. Protective actions would be implemented in these areas beyond the EPZ in an ad hoc manner, which means that such actions would follow the existing local all-hazards emergency response plans.

For dose calculation purposes, evacuees are treated in the model as traveling to a point 30 miles from the site. This treatment is consistent with previous calculations (e.g., NUREG-1150) where evacuees moved 10 miles beyond the evacuation zone, at which point they were assumed to receive no further dose. Previous analyses chose the evacuation zone to represent the EPZ and did not consider shadow evacuation. The SOARCA analyses consider a shadow evacuation beyond the EPZ out to a distance of 20 miles from the plant. Thus, evacuating to a 30-mile radius results in the outermost evacuees traveling 10 miles beyond their initial location.

A shadow evacuation is the voluntary (self-initiated) evacuation of members of the public from areas that are not under official evacuation orders and typically occurs when a large scale evacuation is ordered. Shadow evacuations are often reported and observed, but there is little quantitative data available regarding these evacuations. SOARCA models a shadow evacuation of 20 percent of the public residing in the 10- to 20-mile area beyond the EPZ [48] based on data from a national telephone survey of residents of EPZs. In the survey, about 20 percent of people who had been previously asked to evacuate had also evacuated for situations in which they were asked not to evacuate (e.g., shadow evacuation). The size of the shadow evacuation is greatly affected by the emergency messaging used by the local authorities and could be larger or smaller. A shadow evacuation can delay the evacuation of people closer to the plant increasing their risk of exposure. SOARCA modeled shadow evacuations to improve realism.

The initiating event for many of the accident scenarios considered by SOARCA is a large earthquake close to the plant site. For this event, it was assumed that severe damage would be generally localized (e.g., 30–40 kilometers from the site). The SOARCA team considered the effects of such an earthquake on emergency response capabilities onsite and offsite as well as the evacuation speed of the public. However, considerable uncertainty exists in characterizing the impacts of an earthquake, and the SOARCA project therefore addressed the earthquake effects in a separate analysis. A consequence analysis was performed for the accident sequences for each site, and a single seismic analysis was performed for the more challenging accident postulated for each site.

The study performed a limited and conservative seismic analysis of local infrastructure, which may affect evacuation activities for each site. The seismic analysis indicated that long-span bridges, typically bridges crossing the river and interstate roadway crossings (at Surry), close to each site are unlikely to survive the earthquake and are assumed to be impassable throughout the emergency response. The study also assumed that some smaller bridges and road crossings would fail, as well as some roadways where underlying soils could slide off into adjacent waterways. Residential and commercial structures would be damaged but generally would survive the earthquake. The local electrical grid is assumed to be out of service through automatic shutdown or equipment failures; however, it is not expected that power would be out within the entire 10-mile (314-square-mile) EPZ. A limited backup power system is in place for the sirens at Peach Bottom, allowing some sirens to operate. Backup power is available for the sirens at Surry and those sirens are assumed to operate. OROs would perform route alerting to notify the population of the need to take protective actions in areas where sirens are not functional. Route alerting consists of emergency responders driving through neighborhoods using loudspeakers or going door to door to notify residents of the emergency and is a routine and effective method of informing the public [47]. 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.

5.3.1 Base Case Analyses of Emergency Response

The SOARCA project used WinMACCS to develop and model a case for each accident sequence that resulted in a radioactive release to the environment that would invoke protective actions. Initial protective actions at Surry, for which Supplement 3 to NUREG-0654/FEMA-REP-1, Revision 1 [63] provides guidance, would likely include evacuation of the 2-mile zone around the NPP and of a 5-mile downwind keyhole and would expand to 10 miles, if necessary, based on dose projections. Pennsylvania implements a 360-degree, 10-mile evacuation. For consistency in approach, the analyses for both sites included 360-degree evacuation of the public residing within the 10-mile EPZ. Figure is an example of evacuation routing. In addition, the analyses included a 20 percent shadow evacuation of the public residing in the 10- to 20-mile area beyond the EPZ [48]. The population beyond 20 miles was not assumed to evacuate, although this segment of the population is relocated if projected doses exceed EPA guidelines.

The project established six cohorts for each site, each of which represents a discrete segment of the population that has different response characteristics. The use of six cohorts provides greater fidelity in the treatment of emergency response than was possible for previous studies, because MACCS2 previously allowed a maximum of three cohorts. As a general assumption, the accident scenario was assumed to occur during school hours, and one cohort was established for schoolchildren within the EPZ. Other cohorts included the general public within the EPZ, special facilities within the EPZ, the evacuation tail, shadow evacuees, and a nonevacuating cohort. The nonevacuating cohort represents a small fraction (0.5 percent in this case) of the population who may choose not to evacuate when directed to do so.

The SOARCA project used the evacuation time estimates (ETEs) provided by the licensees to develop speeds for evacuating cohorts. For NPPs, Appendix E to 10 CFR 50.47, "Emergency Plans," requires that licensees provide an analysis of the time required to evacuate transient and permanent residents for various sectors and distances within the EPZ. Developed by licensees to support this requirement, an ETE is a tool that gives emergency managers information on how long it may take to evacuate a portion or all of the EPZ. Using this information, emergency managers can decide if evacuation is the most appropriate protective action for a specific accident. The site-specific ETEs were used to establish the evacuation-related input parameters for MACCS2.

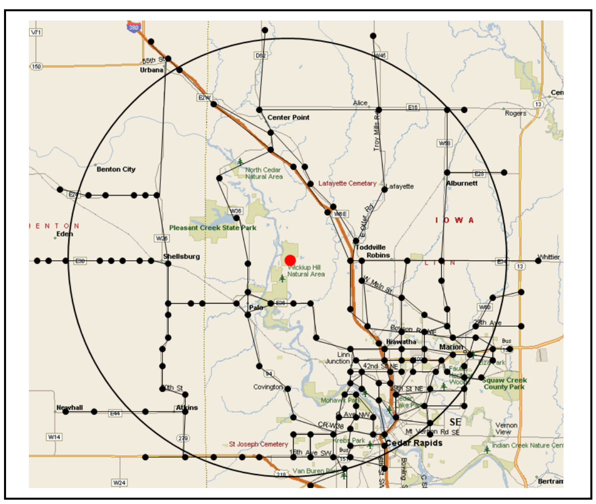


Figure 11 Typical evacuation routing

Using WinMACCS, the parameters related to emergency planning protective actions were input to the consequence modeling. WinMACCS permits temporal and spatial elements of sheltering and evacuation to be modeled and allows for the movement of multiple cohorts and accommodates speed and direction variations for each evacuating cohort. To develop the input, a review of the evacuation routes determined the likely directions in which evacuees would travel. The evacuation area was mapped onto a grid with 64 compass sectors and 15 radii that formed the basis for the network evacuation model. All accident sequences at each site had the same evacuation network. Response timing and evacuation speed parameters were created specifically for each accident sequence. A newly developed option in WinMACCS and MACCS2 allowed an adjustment within grid elements to increase or decrease speeds based on expected traffic congestion. In addition, for cases where the hourly meteorological data included precipitation, the speeds of all evacuating cohorts were reduced. For dose modeling purposes, all evacuees were assumed to travel to a point 30 miles from the site. This distance accounts for the fact that the assembly sites are at some distance from the plant. Whether doses are actually received by the evacuees during any part of their travel is a complicated function of the direction of travel and the times and the directions of the plumes as they are released from the plant. Each plume disperses in a straight line in its own downwind direction.

5.3.2 Sensitivity Analyses of Emergency Response

After completion of the base case analysis, the following three variations were conducted as sensitivity analyses:

(1) <u>Evacuation to a distance of 16 miles from the plant</u>. This analysis assessed the complete evacuation of 16 miles around the plant. It assumed the members of the public in the 16- to 20-mile zone would shelter.

(2) <u>Evacuation to a distance of 20 miles from the plant</u>. This analysis developed an ETE for the 20-mile area to provide realistic modeling parameters for the movement of the public.

(3) <u>Delay in implementing protective actions</u>. This analysis included an assumption that there could be a 30-minute delay in implementing protective actions by the public. This sensitivity study assumes that a delay could occur in notification to offsite authorities, notification from offsite authorities to the public, receipt of the warning by the public, or for other reasons. The analysis assumed that cohorts take 30 minutes longer to start implementing protective actions than in the base case analysis.

The first two sensitivity studies used the Oak Ridge Evacuation Modeling System to create additional ETEs for each site to establish speeds and mobilization parameters for movement of the public residing between 10 and 20 miles. This system developed ETEs for the general public within the 10- to 20-mile zone.

The development of the parameters for the sensitivity analyses showed that, for the larger evacuation areas, the travel speeds were typically slower than in the baseline analyses. This was because of the additional vehicle load on the roadway network in the more populated areas, such as Lancaster, PA. For the third sensitivity analysis, which assessed a delay in implementation, the speeds remained unchanged. The effects of these variations in emergency response on latent cancer fatality risk calculations are discussed further in Section 6.5.

5.4 Source Term Evaluation

A source term evaluation for each of the accident scenarios used MELMACCS [49], which reads a MELCOR plot file and extracts information useful for source term definition for MACCS2. MELMACCS requires the selection of a number of user options. The following paragraphs describe the specific choices made for SOARCA.

The first set of choices is related to the chemical groups or classes to be included in the analysis. Here, the analyses included the standard set of fission product groups (i.e., the xenon, cesium, barium, iodine, tellurium, ruthenium, molybdenum, cerium, and lanthanum groups). A related quantity defining the burnup to be assumed when calculating the fission product inventory depends on the plant type. In an effort to provide a best estimate fission product inventory for Peach Bottom, SOARCA used an ORIGEN calculation to estimate the inventory at midcycle for which peak rod burnup is estimated to be 49 MWd/kg. MELMACCS used these data to specify the inventory for MACCS2, and the MACCS2 input is, therefore, consistent with the MELCOR calculation. Surry used a previously available fission product inventory based on the regulatory limit of burnup (65 MWd/kg for the peak fuel rod). This inventory is conservative.

A set of parameters define the ground elevation (grade) in the MELCOR reference frame, the height of the building from which release occurs, and the initial plume dimensions. The SOARCA MELCOR analyses use reactor shutdown as the reference time, so the time of accident initiation is always set to zero in the MELMACCS input.

MELMACCS calculates aerosol deposition velocities based on the geometric mean diameter of each aerosol bin as defined in the MELCOR analysis. Expert elicitation data form the basis for the deposition velocities, using the median value of the combined distribution from the experts [50]. This report applies the MELMACCS equation to determine the deposition velocities. The equation accounts for the dependence of deposition velocity on surface roughness, wind speed, and aerodynamic particle diameter. MELMACCS also accounts for independent particle size distributions for each of the chemical groups reported by MELCOR. Thus, the aerosol size distribution and the resulting deposition velocity distribution are generally different for each chemical group. This is accounted for by assigning different mass fraction distributions across the aerosol bins for each chemical group in the MACCS2 input.

Typical values for surface roughness and mean wind speed, 0.1 m and 2.2 m/s, respectively, are used as inputs in MELMACCS. This value of surface roughness is commonly used for consequence analyses in the United States and is consistent with the value in NUREG-1150 [2]. However, a value that is more representative of the terrain at the two sites is also investigated as a sensitivity study and the results documented in the reports for the two sites. The specific weather files determined the mean wind speeds used in the consequence analyses. Table 3 displays the deposition velocities in SOARCA analyses for both Peach Bottom and Surry.

Table 3 Deposition Velocities Used in the SOARCA Analyses

Bin #	Median Diameter (µm)	Deposition Velocity (m/s)
1	0.15	5.35×10^{-4}
2	0.29	4.91×10^{-4}
3	0.53	6.43×10^{-4}
4	0.99	1.08×10^{-3}
5	1.8	2.12×10^{-3}
6	3.4	4.34×10^{-3}
7	6.4	8.37×10^{-3}
8	11.9	1.37×10^{-2}
9	22.1	1.70×10^{-2}
10	41.2	1.70×10^{-2}

MELCOR results include the relative quantities of aerosols contained in each size bin listed in the table. MACCS2 uses this information, plus the deposition velocities in the table, to determine the rate of depletion of aerosols from the plume. Generally, the larger aerosols deposit more quickly and so are depleted more rapidly from the plume. The peak in the aerosol size distribution is usually a few micrometers **(**µm**)**, which corresponds to a deposition velocity of a few millimeters per second.

Finally, significant releases were broken up into 1-hour plume segments. MACCS2 allows plume segments to travel in only one compass direction based on weather data. More plume segments can better represent plume transport and dispersion caused by possible changes in the weather (such as the wind direction) during the release. Longer plume segments were sometimes used for trivial releases, such as those where the segment content is a very small fraction of the total release. Finer resolution of these releases was not necessary to maintain the fidelity of the calculation. The MELCOR analyses provided the amount of each chemical element group in each aerosol bin for each plume segment.

5.5 Site-Specific Parameters

The SOARCA project took weather data for each site from meteorological archives provided by each plant (see Section 5.2). It then processed the raw data into 64 compass sectors to use the angular resolution capabilities in WinMACCS 3.6 and MACCS2 2.5.

SECPOP2000 [51] initially created site files for 16 compass sectors, which is the only angular resolution supported by that code. WinMACCS then interpolated these site files onto the 64-compass-sector grid used for the consequence analyses. The granularity of the population data for 16 compass directions is maintained for the 64 compass directions data. The SECPOP2000 population data were also scaled by a factor of 1.0533 to account for the average U.S. population growth between the years 2000 and 2005. 2010 U.S. Census data was not used because most calculations were already completed by the time it was released. Changes in population over the last decade are not expected to have a significant impact on any of the calculated individual latent cancer fatality risks reported in Chapter 6.

Consequence analyses used the standard approach of evaluating accidents in the following two phases:

(1) Emergency phase. This phase begins with the initiating event and continues for about 1 week. The release from the plant and plume transport through the MACCS2 grid occurs during this phase. This phase also includes emergency response (i.e., evacuation and relocation of the population to reduce exposures and doses). The project chose the length of this phase to ensure that all plumes can exit the calculational mesh during the period, because certain assumptions about doses (e.g., that all late phase doses are small enough to warrant applying the dose and dose rate effectiveness factor) could be questionable if the early phase was made too short.

(2) Long-term phase. This phase is the period following the emergency phase and continues for 50 years. Three actions take place during the long-term phase. Land that is contaminated above the level that is allowable for habitation is decontaminated and potentially interdicted for an additional period. During this time, the land is not available for human habitation. Land that cannot be restored to habitability is condemned, in which case the residents do not return during the long-term phase.

Both sites needed a choice for surface roughness, which affects both vertical dispersion and deposition velocities. A generic value of 10 centimeters for surface roughness was selected for the consequence analyses, just as it had been in NUREG-1150 and most other previous studies. However, sensitivity studies also evaluated the effect of site-specific surface roughness. The reports for each of the sites document these sensitivity studies.

The project evaluated shielding factors applied to evacuation, normal activity, and sheltering for each relevant dose pathway (i.e., inhalation, deposition onto skin, cloudshine, and groundshine) for each site based on values used in NUREG-1150 [2] and NUREG-6953, Volume 1, "Review of NUREG-0654, Supplement 3, 'Criteria for Protective Action Recommendations for Severe Accidents,'" issued December 2007 [52]. A review of the discussion of shielding in the NUREG-1150 documentation suggests that the factors the authors considered were adequate for SOARCA purposes. One departure from the NUREG-1150 values is for normal activity. The SOARCA project reevaluated each of the normal activity values, assuming that the average person spends 19 percent of the day outdoors and 81 percent of the day indoors [63]. It evaluated the value for each of the pathways as a linear combination of 19 percent of the value for evacuation and 81 percent of the value for sheltering.

For dose calculations, the project modeled evacuees as traveling to a distance of 30 miles from the plant. In addition, it relocated the nonevacuating cohort and the public beyond the EPZ from areas where the projected dose during the emergency phase exceeded a set of two upper bounds. These bounds were based on dose levels published by EPA, which are 1 to 5 rem. SOARCA used the upper limit of this range (5 rem) to trigger hot-spot relocation for both Surry and Peach Bottom and used the lower limit of this range (1 rem) to trigger normal relocation for Surry, while it used 0.5 rem for Peach Bottom, to be consistent with the Pennsylvania habitability criterion.

MACCS2 performs hot-spot relocation first and normal relocation second. The choices of times associated with normal and hot-spot relocation depended on the specific accident scenario, because they are based on plume arrival. The scenario-specific time for completion of the relocation includes the time for response personnel to identify the involved area, for them to notify the residents within that area that relocation is necessary, and for the residents to remove themselves from the area. Because the timing of relocation is keyed to plume arrival, there is always a period of exposure before initiation of relocation. Volumes 1 and 2 of NUREG/CR-7110, discuss the specific choices for the parameters controlling the exposure period.

Site-specific values determine long-term habitability. Most States adhere to EPA guidelines that allow a dose of 2 rem in the first year and 500 millirem (mrem) per year thereafter. The EPA recommendation that has traditionally been implemented in MACCS2 is 4 rem over 5 years (2 rem in the first year + 4 years × 0.5 rem) of exposure, and this study adopts that convention. MACCS2 cannot explicitly use the EPA recommendation because MACCS2 accepts only one dose and one time period. Some States, like Pennsylvania, have a stricter habitability criterion (i.e., 0.5 rem/year beginning in the first year). Thus, the habitability or return criterion is site specific, as Volumes 1 and 2 of NUREG/CR-7110 discuss further.

Some States have distributed potassium iodide (KI) tablets to people who live near commercial NPPs. KI has been distributed within the EPZ at the Peach Bottom and Surry sites. The purpose of the KI is to saturate the thyroid gland with iodine so that further uptake of iodine by the thyroid is diminished. If taken at the right time, KI can nearly eliminate doses to the thyroid gland from inhaled radioiodine. Ingestion of KI is modeled for half of the residents near plants where KI has been distributed by the State or local government. A further assumption is that most residents do not take KI at the optimal time (from shortly before to immediately after plume arrival), so the efficacy is only 70 percent (i.e., the thyroid dose from inhaled radioiodine is reduced by 70 percent).

Other site-specific parameters include farmland and nonfarmland values. These values are also scaled from NUREG-1150 values using the Consumer Price Index as the basis for price escalation. A scaling factor of 1.09 accounts for inflation between the years 2002 and 2005. Land values influence the decision to decontaminate, interdict, or condemn land. If the assessed cost of decontamination is higher than the land value, the assumption is the land would be condemned. Because the public would not be allowed to return to condemned land and, therefore, no dose would be received, the land values did have a second-order effect on the predicted long-term health consequences.

5.6 Non-Site-Specific Parameters

The SOARCA analyses do not treat the ingestion of contaminated food and water. The reasoning is that abundant supplies of food and water are available in the United States and can be distributed to areas affected by a reactor accident.

Much of the non-site-specific data used for consequence analysis in SOARCA are taken from reports that document a joint NRC/Commission of the European Communities expert elicitation study [50]. These data include atmospheric dispersion parameters, dry deposition velocities, wet deposition parameters, and acute health-effect parameters. In all cases, the point-value consequence analyses in SOARCA use median values extracted from the elicitation study [50].

The SOARCA analyses based the dose conversion factors on Federal Guidance Report (FGR) -13, "Cancer Risk Coefficients for Environmental Exposure to Radionuclides: Updates and Supplements," CD Supplement, Revision 1, issued April 2002 [53]. This guidance report also recommended changes to the biological effectiveness factors (BEFs) for alpha radiation for two of the organs used to estimate latent cancer health effects, to be consistent with the evaluation of risk factors for cancers associated with those organs. The two organs are bone marrow and breast; for these organs, the BEFs for alpha radiation were changed from the standard value of 20 to 1 and 10, respectively. Doses to these organs are used to evaluate occurrences of leukemia and breast cancer, respectively. The choice of BEFs for these tissues is dictated by EPA 402-R-93-076, "Estimating Radiogenic Cancer Risks," issued June 1994 [54].

A May 2008 ORNL memo, "Risk Coefficients for SOARCA Project" [55], also recommended using dose to the pancreas as a surrogate for dose to soft tissue to estimate residual cancers. The reason for the choice of the pancreas is that it is not a tissue in which inhaled material is deposited. Because MACCS2 does not currently read the data for the pancreas from the dose conversion factor file, a workaround was created. Values of the dose coefficients for the pancreas were copied into the organ called bladder wall. Thus, residual cancers are associated with the organ called bladder wall, which actually contains data for the pancreas.

The SOARCA study applied a dose and dose rate effectiveness factor to all doses in the late phase of the offsite consequence calculation and to those doses in the early phase that were less than 20 rem to the whole body. This factor, which appears in the denominator, accounts for the fact that protracted low doses are perceived to be less effective in causing cancer than acute doses. The dose and dose rate effectiveness factor for all cancers except for the breast was 2.0, and for the breast, it was 1.0.

The ORNL memorandum [55] also recommended risk factors for latent health effects that come from the National Research Council's Committee on the Biological Effects of Ionizing Radiations (BEIR) V report, "Health Effects of Exposure to Low Levels of Ionizing Radiation," issued 1990 [56], and are consistent with the modified dose conversion factor file described in the preceding paragraph. These risk factors include seven organ-specific cancers plus residual cancers that are not accounted for directly. In 2009, the National Research Council released the BEIR VII report, an additional study of the biological effects of ionizing radiation. No one-to-one correspondence exists between the cancers reported in BEIR VII and those in the earlier BEIR V report. Therefore, the dose coefficients of tissues of the body in FGR-13 may or may not be consistent with the BEIR VII cancer sites. Thus, the SOARCA staff decided to await EPA's review of BEIR VII and subsequent update of FGR-13 before implementing BEIR VII risk coefficients.

Values from NUREG-1150 [2] provide the basis for decontamination parameters, which consist of two levels of decontamination, just as in NUREG-1150. The cost parameters associated with decontamination are adjusted to account for inflation using the Consumer Price Index. This report does not consider costs associated with a reactor accident; however, these parameters do affect decisions on whether contaminated areas can be restored to habitability and therefore affect predicted doses and risk of health effects.

5.7 Estimating Latent Cancer Fatality Health Effects

Experts generally agree that it is difficult to characterize cancer risk because of the low statistical precision associated with relatively small numbers of excess cases at low doses. This limits the ability to estimate trends in risk. From an epidemiological standpoint, the number of LCFs attributable to radiation exposure from accidental releases from a severe accident would not be statistically detectable above the normal rate of cancer fatalities in the exposed population (i.e., the excess cancer fatalities predicted are too few to allow the detection of a statistically significant difference in the cancer fatalities expected from other causes among the same population). For example, in 2006, the World Health Organization estimated that 16,000 European cancer deaths would be attributable to radiation released from the 1986 Chernobyl NPP accident, but these predicted numbers are small relative to the several hundred million cancer cases that are expected in Europe through 2065 from other causes. Moreover, the World Health Organization concluded that "it is unlikely that the cancer burden from the largest radiological accident to date could be detected by monitoring national cancer statistics."

New findings have been published from analyses of fractionated or chronic low-dose exposure to low, linear energy transfer radiation. In particular, these recent findings included a study of nuclear workers in 15 countries, studies of persons living in the vicinity of the Techa River in the Russian Federation who were exposed to radioactive waste discharges from the Mayak Production Association, a study of persons exposed to fallout from the Semipalatinsk nuclear test site in Kazakhstan, and studies in regions with high natural background levels of radiation. Cancer risk estimates in these studies are generally derived from the Japanese atomic bomb data. The most recent results from analyzing these data are consistent with a linear or linear-quadratic dose-response relationship of all solid cancers together and with a linear-quadratic dose-response relationship for leukemia. A linear-quadratic form for a dose model has a dependence on the square of the dose, as well as on the dose itself.

In the absence of additional information, the International Commission on Radiological Protection (ICRP), the National Academy of Sciences, and the United Nations Scientific Committee on the Effects of Atomic Radiation have each indicated that the current scientific evidence is consistent with the hypothesis that an LNT dose-response relationship exists between exposure to ionizing radiation and the development of cancer in humans.

Conversely, in *"Dose-effect relationships and estimation of the carcinogenic effects of low doses of ionizing radiation,"* dated March 30, 2005 [57], the French National Academy of Medicine advocates the following on page 1:

> A linear no-threshold relationship (LNT) describes well the relation between the dose and the carcinogenic effect in this dose range (0.2 to 3 Sv) [to the whole body] where it could be tested. However, the use of this relationship to assess by extrapolation the risk of low and very low doses deserves great caution. Recent radiobiological data undermine the validity of estimations based on LNT in the range of doses lower than a few dozen mSv which leads to the questioning of the hypotheses on which LNT is implicitly based.

Although the French National Academy of Medicine raises doubts about the validity of using LNT to evaluate the carcinogenic risk of low doses (less than 100 millisieverts (mSv) (10 rem)), and particularly for very low doses (less than 10 mSv (1 rem)), it did not articulate the exact value that should be ascribed to a dose threshold.

Ultimately, MACCS2 converts external and internal exposures to individual members of the public from collective organ dose to LCFs. The LNT model raises the concern that the summation of very small exposures may inappropriately attribute LCFs to individuals receiving these exposures. Organizations such as ICRP and the Health Physics Society (HPS) consider it to be an inappropriate use of these exposures. While the National Council on Radiation Protection and Measurements (NCRP) supports the LNT model, it recommends binning exposures into ranges and considering those ranges separately. Moreover, in situations involving very small exposures to large populations, ICRP and NCRP have noted that the most likely number of excess health effects is zero when the collective dose to such populations is equivalent to the reciprocal of the risk coefficient (about 20 person-sieverts (Sv) (2,000 person-rem)). Nevertheless, issues remain related to assessing public exposure, estimating offsite consequences, and communicating these assessments to the public. Several organizations such as ICRP have addressed this issue. In its most recent recommendations (ICRP Report 103, "The 2007 Recommendations of the International Commission on Radiological Protection," approved March 2007 [58]), ICRP stated the following:

> Collective effective dose is an instrument for optimization, for comparing radiological technologies and protection procedures. Collective effective dose is not intended as a tool for epidemiological studies, and it is inappropriate to use it in risk projections. This is because the assumptions implicit in the calculation of collective effective dose (e.g., when applying the LNT model) conceal large biological and statistical uncertainties. Specifically, the computation of cancer deaths based on collective effective doses involving trivial exposures to large populations is not reasonable and should be avoided. Such computations based on collective effective dose were never intended, are biologically and statistically very uncertain, presuppose a number of caveats that tend not to be repeated when estimates are quoted out of context, and are an incorrect use of this protection quantity.

Although ICRP provided qualitative guidance on situations where collective dose should not be used, it did not provide guidance on when these concepts actually are, and are not, appropriate, nor did it clearly articulate the boundaries within which the calculations are valid, as well as the dose ranges for which epidemiological and cellular or molecular data provide information on the health effects associated with radiation exposure. ICRP did note, however, that when ranges of exposures are large, collective dose may aggregate information inappropriately and could be misleading for selecting protective actions.

The National Academy of Sciences reported the following [56]:

> The magnitude of estimated risk for total cancer mortality or leukemia has not changed greatly from estimates in past reports such as Biological Effects of Ionizing Radiation (BEIR) and recent reports of the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) and ICRP. New data and analyses have reduced sampling uncertainty, but uncertainties related to estimating risk for exposure to low doses and dose rates and to transporting risks from Japanese A-bomb survivors to the U.S. population remain large.

The National Academy of Sciences goes on to conclude that "current scientific evidence is consistent with the hypothesis that there is a linear, no-threshold dose-response relationship between exposure to ionizing radiation and the development of cancer in humans."

Many groups acknowledge the uncertainties associated with estimating risk for exposure to low radiation doses. One important question that remains is what health consequences, if any, are attributable to very low radiation exposure. In its most recent recommendations (ICRP Report 103 [58]) described above, ICRP warned that the computation of cancer deaths based on collective effective doses involving trivial exposures is not reasonable and should be avoided. However, the report did not explicitly provide a quantitative range for which exposures should not be considered. However, in its 2007 Report 104, "Scope of Radiological Protection Control Measures" [59], ICRP concludes that the radiation dose that is of no significance to individuals should be in the range of 20–100 microsieverts (μSv) (2–10 mrem) per year whole body dose. The International Atomic Energy Agency has stated that an individual dose is likely to be regarded as trivial if it is on the order of some several millirem per year.

Alternatively, HPS developed a position paper, "Radiation Risk in Perspective," revised August 2004 [60], to specifically address quantitative estimation of health risks. This position paper concludes that quantitative estimates of risk should be limited to individuals receiving a whole body dose greater than 0.05 Sv (5 rem) in 1 year or a lifetime dose greater than 0.1 Sv (10 rem) in addition to natural background radiation. HPS also concluded that risk estimates should not be conducted below these doses. The position paper further states that low dose expressions of risk should be only qualitative, discuss a range of possible outcomes, and emphasize the inability to detect any increased health detriment.

The LNT model provides a viewpoint that is consistent with the NRC regulatory approach, and past analyses using the MACCS2 code have assumed an LNT dose-response model. The NRC is neither changing nor contemplating changing radiation protection standards and policy as a result of an approach taken in the SOARCA study to characterize offsite health consequences for low probability events. Still, the NRC can use different approaches for different applications. Therefore, the SOARCA analyses consider a range of dose truncation values ranging from LNT to a dose truncation level based on the HPS position that there is a dose below which, because of uncertainties, a quantified risk should not be assigned, which is 5 rem/year with a lifetime dose limit of 10 rem.

The SOARCA analyses also considered two additional dose truncation levels. One is the 10 mrem/year dose truncation value suggested in ICRP Report 104 [59]; the other is U.S. average background radiation combined with average annual medical exposure as a dose truncation level (abbreviated as US BGR), which is 620 mrem/year. Results for three of these four dose truncation levels are reported for each of the accident scenarios considered in the SOARCA study. The results for the 10-mrem/year dose truncation levels were calculated but are not included in the report because the results are very similar to LNT and are also always slightly less than the LNT results.

5.8 Risk Metrics Reported

The statistic that is chosen to convey the likelihood of LCFs resulting from an accident at an NPP is the mean, population-weighted individual risk. This value is more meaningful than the predicted number of LCFs in the sense that it may be compared with cancer fatality rates that have other causes. Individual risks can be presented as conditional risks (i.e., as if the accident had taken place) or as absolute risks (i.e., accounting for the likelihood of the accident occurring per year of reactor operation). The latter definition of risk is more useful, because it conveys the full meaning of risk, which is probability (or frequency) times consequence.

The term "population-weighted" in the preceding paragraph carries the meaning of the effect of population distribution along with wind-rose probabilities of 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 weighted averages over the annual weather trials used in the analysis. The work presented in this report considers uncertainty in the weather. Subsequent work will explore the effect on the predictions of uncertainties in other input parameters.

The mean, population-weighted individual risks range from 0 to 50 miles. The 0- to 10-mile range represents the population within the EPZ. Analyses of severe accident mitigation and severe accident mitigation design alternatives generally use the range from 0 to 50 miles.

6.0 RESULTS AND CONCLUSIONS

To assess the benefits of the various mitigative measures and to provide a basis for comparison to past analyses of unmitigated severe accident scenarios, the SOARCA project treated the selected scenarios in two separate and distinct manners. In the first, it analyzed scenarios that included an assessment of reasonable mitigation measures for which procedures and equipment (and training) exist. Alternatively, if adequate time exists, this would suffice for implementation in lieu of fully developed procedures and training (especially for simple actions), e.g., refilling water storage capabilities. In the second manner, it assumed that the key or vital measures necessary to prevent core damage or to mitigate radiological release were not taken to compare them with previous analyses of unmitigated scenarios. This comparison could reveal the benefits of improved severe accident phenomenological understanding and modeling.

6.1 Mitigation

The SOARCA assessment and analyses demonstrate the feasibility and potential benefits of 10 CFR 50.54(hh) mitigation for the analyzed scenarios. This was demonstrated by an assessment of mitigation (e.g., plant walkdowns and reviews of procedures, available time for implementation, and potential seismic impacts) and the MELCOR calculations, which assess core cooling and the delay or reduction of release. The security-related measures to provide alternative ac power and portable diesel-driven pumps were especially helpful in counteracting SBO scenarios. For the ISLOCA scenario, installed equipment was adequate to prevent core damage owing to the time available for corrective action. For all events except one, the mitigation was sufficient to prevent core damage. For one event, the Surry STSBO and a variant of the STSBO, in which a thermally-induced steam generator tube rupture also occurs, the mitigation was sufficient to enable flooding of the containment through the containment spray system to cover core debris resulting from vessel failure. The assessment of the mitigation measures was supported by the integrated accident progression analyses using the MELCOR code. MELCOR analyses both confirmed the time available to take mitigation measures and confirmed that those measures, once taken, were adequate to prevent core damage or significantly reduce radiological releases. In other instances, MELCOR analyses using only installed equipment revealed that PRA success criteria were overly conservative, indicating core damage, where MELCOR analysis indicated no core damage. Mitigation results are included in Tables 4 through 7.

6.2 Accident Progression and Radionuclide Release

An important result of the MELCOR analyses was that the select severe accidents proceed much more slowly than the SST1 case from the 1982 Siting Study. The reasons for this are threefold: (1) research and development of better phenomenological modeling has produced a much more protracted and delayed core degradation transient with substantial delays of reactor vessel failure, (2) all aspects of accident scenarios receive more realistic treatment, which includes more complete modeling of plant systems and often yields delays in core damage and radiological release, and (3) the scope of SOARCA focuses on the more likely and important accident scenarios, while past treatments included less likely accident progressions. In general, the bounding approaches in past simplified treatments used qualitative logical models. In

SOARCA, where specific self-consistent scenarios are analyzed in an integral fashion using MELCOR, the result is that accident conditions or attributes that contribute to a more severe response in one area may produce an ameliorating effect in another area.

For the LTSBO scenarios for both Peach Bottom and Surry (the most likely severe accident scenario for each plant considered in SOARCA) analyzed assuming no mitigation, core damage begins in 9 to 16 hours, and reactor vessel failure begins at about 20 hours. Offsite radiological release due to containment failure begins at about 20 hours for Peach Bottom (BWR) and at 45 hours for Surry (PWR). The SOARCA analyses therefore show that time may be available for operators to take corrective action and get additional assistance from plant technical support centers even if initial efforts are assumed unsuccessful. For the most rapid events (i.e., the unmitigated STSBO in which core damage may begin in 1 to 3 hours), reactor vessel failure begins at roughly 8 hours, possibly allowing time to restore core cooling and prevent vessel failure. In these cases, containment failure and radiological release begins at about 8 hours for Peach Bottom and at 25 hours for Surry. For the unmitigated Surry ISLOCA, the offsite radiological release begins at about 13 hours and in the other bypass event analyzed, the TISGTR, the radiological release begins at about 3.5 hours but is shown by analyses to be substantially smaller than the 1982 Siting Study SST1 release.

Table 4 and Table 5 provide key accident progression timing results for SOARCA scenarios. Table 4 shows the same times for lower head failure and start of the release to the environment, because drywell shell melt-through occurs about 15 minutes after lower head failure.

Table 4 Peach Bottom Accident Progression Timing Results

Scenario	Mitigated			Unmitigated		
	Time to start of core damage (hours)	Time to lower head failure (hours)	Time to start of release to environment (hours)	Time to start of core damage (hours)	Time to lower head failure (hours)	Time to start of release to environment (hours)
Long-term SBO	No Core Damage			9	20	20
Short-term SBO	No Core Damage*			1	8	8
Short-term SBO with RCIC Blackstart				7	17	17

* If the RCIC blackrun was successfully controlled, then the mitigated STSBO with RCIC blackstart would be functionally similar to the mitigated LTSBO (i.e., no core damage). This was qualitatively determined based on the timing and equipment availabilities from the other SBO analyses.

Table 5 Surry Accident Progression Timing Results

Scenario	Mitigated			Unmitigated		
	Time to start of core damage (hours)	Time to lower head failure (hours)	Time to start of release to environment (hours)	Time to start of core damage (hours)	Time to lower head failure (hours)	Time to start of release to environment (hours)
Long-term SBO	No Core Damage			16	21	45
Short-term SBO	3	7	66	3	7	25
Thermally induced steam generator tube rupture	3	7.5	3.5*	3	7.5	3.5
Interfacing systems LOCA	No Core Damage			13	19	13

* Although the time at which release to the environment starts is the same in the mitigated and unmitigated cases, containment failure is delayed by about 46 hours in the mitigated case compared to the unmitigated case.

The SOARCA study also demonstrated that the magnitude of the environmental radionuclide release is likely to be much smaller than the SST1 source term, again as a result of (1) extensive research and improved modeling and (2) integrated and more complete plant simulation. Historically important radionuclides have included the more volatile fission products (i.e., those released in greater quantity from the overheated fuel) such as iodine and cesium. These two radionuclides have also been useful representatives of radionuclides with a short half-life (iodine) and those with a long half-life (cesium). SOARCA analysis typically predicts iodine releases on the order of 1-2 percent for the dominant scenarios with the highest releases on the order of 10-15 percent for the lower frequency, more severe scenarios. By contrast, the SST1 source term in the 1982 Siting Study assumed an iodine release of 45 percent. With respect to cesium, SOARCA predicts releases of 2 percent or less. By contrast, the SST1 source term assumed a cesium release of 67 percent. Figure and Figure provide the radionuclide release results for iodine and cesium.

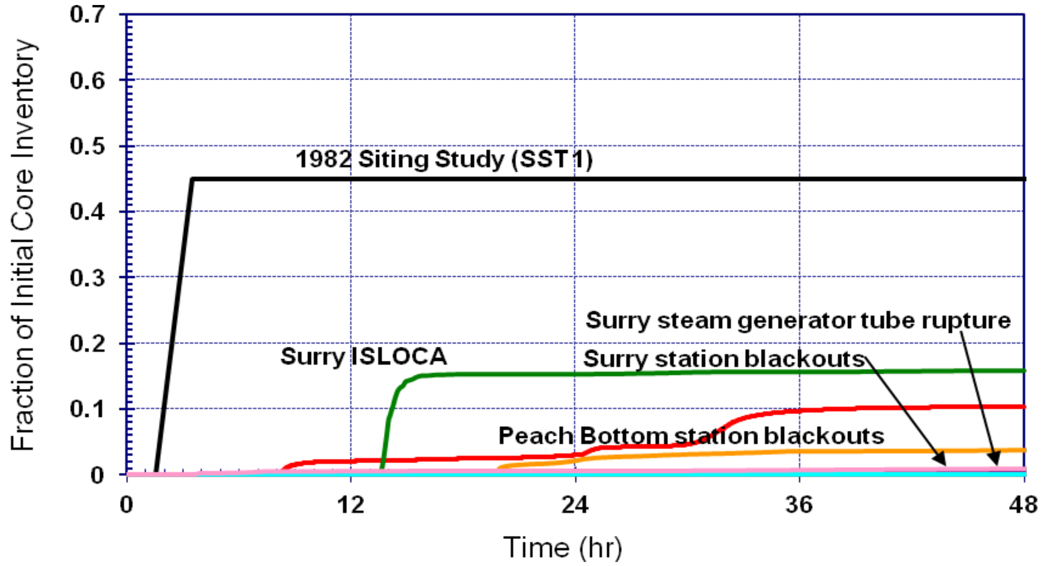


Figure 12 Iodine releases to the environment for SOARCA unmitigated scenarios and the 1982 Siting Study SST1 case

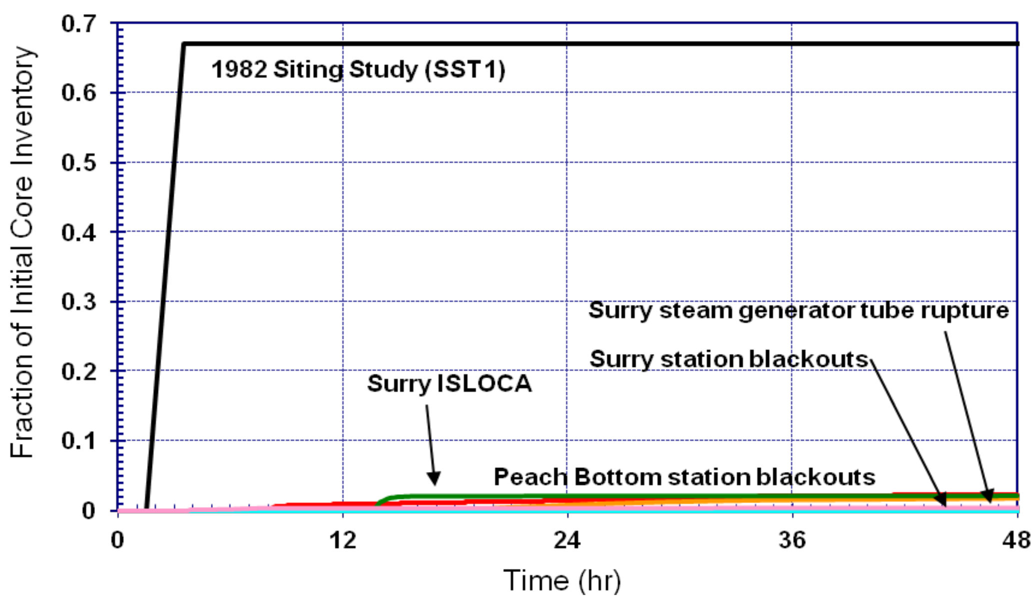


Figure 13 Cesium releases to the environment for SOARCA unmitigated scenarios and the 1982 Siting Study SST1 case

Past PRAs and consequence studies showed that sequences involving large early releases were important risk contributors. For example, the 1982 Siting Study SST1 results were controlled by an internally initiated event with a large early release that was assigned a representative frequency of 1×10^{-5} per reactor-year. However, in the SOARCA study, no sequences resulted in a large early release, even considering external events and unsuccessful mitigation. This is a result of research conducted over the last several decades that has shown that phenomena earlier believed to lead to a large early release are of extremely low probability or not physically feasible. This research was focused on phenomena that had been previously assumed to be prime contributors to severe accident risk, including direct containment heating and alpha mode failure.

The PWR SBO with a TISGTR was historically believed to result in a large, relatively early release, potentially leading to higher offsite consequences. However, MELCOR analysis of Surry performed for SOARCA shows that the release is small, because other reactor coolant system piping inside containment (i.e., hot let nozzle) fails soon after the tube rupture and thereby retains fission products within the containment. Also, the release was somewhat delayed; for the STSBO where loss of injection occurred at the start of the accident, the tube rupture and release began about 3.5 hours into the event. Moreover, core damage, tube rupture, and radiological release could be delayed for many hours if auxiliary feedwater were available even for a relatively short time.

6.3 Offsite Radiological Consequences

The result of the accident progression and source term analysis is that releases are delayed, smaller, and more dispersed relative to the 1982 Siting Study SST1 case. This fact, combined with the realistic simulation of emergency response and the greater distances radioactive material is expected to disperse, led to essentially no risk of early fatalities being calculated as close-in populations were evacuated before or shortly after plume arrival.

Because of the last factor, significantly more of the latent cancer fatality risk in the selected SOARCA scenarios comes from low doses compared to the results of the SST1 source term from the 1982 Siting Study. Therefore, a dose truncation significantly reduces the quantified LCF risk in the SOARCA scenarios, much more so than a dose truncation would have for the SST1 from the 1982 Siting Study.

Latent health effects calculated using any of the dose-response models (in combination with the frequency of release) referenced in this study are small in comparison to the NRC Safety Goal. Much of the LCF risk was in fact derived from the small doses received by populations returning to their homes in accordance with emergency planning guidelines. Because much of the health risk is caused by the return of the population, it is therefore controllable. For example, for the Peach Bottom LTSBO, for individuals living within the EPZ, 99 percent of the LCF risk derives from the long-term dose received by the population returning to their homes and being exposed to small radiation doses. Similarly, about 70 percent of the LCF risk to individuals within 50 miles is from returning home. The percentage is larger for the EPZ, because of its evacuation before the start of the release. Here, the calculation of scenario-specific LCF risk, though very small, is strongly influenced by the relationship between low-dose health effects modeling and criteria for allowing the population to return.

Tables 6 and 7 show estimates of conditional (i.e., assuming the accident has occurred) scenario-specific probabilities of an LCF range from roughly 10^{-4} to 10^{-5}, using the LNT dose-response model (other dose models result in lower or much lower conditional risk). The tables also provide the product of this value and the scenario CDF, which is best described as the scenario-specific risk of LCF for an individual located within 10 miles of the plant. Scenario-specific risk of an LCF for an individual within 10 miles of the plant is on the order of 10^{-9} to 10^{-11} per reactor-year. These risk estimates are millions of times lower than the general risk of a cancer fatality in the United States from all causes, approximately 2×10^{-3} per year and thousands of times lower than the NRC Safety Goal.

Comparisons of SOARCA's calculated LCF risks to the NRC Safety Goal [72] and the average annual U.S. cancer fatality risk from all causes [73] are provided to give context that may help the reader to understand the contribution to cancer risks from these nuclear power plant accident scenarios. However, such comparisons have limitations for which the reader should be aware. Relative to the safety goal comparison, the safety goal is intended to encompass all accident scenarios. SOARCA does not examine all scenarios typically considered in a PRA, even though it includes the important scenarios. In fact, any analytical technique, including PRAs, will have inherent limitations of scope and method. As a result, comparison of SOARCA's scenario-specific calculated LCF risks to the NRC Safety Goal is necessarily incomplete. However, it is intended to show that adding multiple scenarios' low risk results in the ~ 10^{-10} range to approximate a summary risk from all scenarios, would yield a summary result that is also below the NRC Safety Goal of $2\text{x}10^{-6}$ or two in one million.

Relative to the U.S. average individual risk of a cancer fatality comparison, the sources of an individual's cancer risk include a complex combination of age, genetics, lifestyle choices, and other environmental factors whereas the consequences from a severe accident at a nuclear plant are involuntary and unlikely to be experienced by most individuals.

Table 6 Peach Bottom Results for Scenarios Assuming LNT Dose-Response Model

<table>
<tr><th rowspan="2">Scenario</th><th rowspan="2">Core damage frequency (CDF) (per reactor-year)</th><th colspan="2">Mitigated</th><th colspan="2">Unmitigated</th></tr>
<tr><th>Conditional scenario-specific probability of latent cancer fatality for an individual located within 10 miles</th><th>Scenario-specific risk (CDF x Conditional) of latent cancer fatality for an individual located within 10 miles (per reactor-year)</th><th>Conditional scenario-specific probability of latent cancer fatality for an individual located within 10 miles</th><th>Scenario-specific risk (CDF x Conditional) of latent cancer fatality for an individual located within 10 miles (per reactor-year)</th></tr>
<tr><td>Long-term SBO</td><td>3×10^{-6}</td><td colspan="2">No Core Damage</td><td>9×10^{-5}</td><td>~ 3×10^{-10} ***</td></tr>
<tr><td>Short-term SBO</td><td>3×10^{-7}</td><td colspan="2" rowspan="2">No Core Damage **</td><td>2×10^{-4}</td><td>~ 6×10^{-11} ***</td></tr>
<tr><td>Short-term SBO with RCIC Blackstart*</td><td>3×10^{-7}</td><td>7×10^{-5}</td><td>~ 2×10^{-11} ***</td></tr>
</table>

* Blackstart of the reactor core isolation cooling (RCIC) system refers to starting RCIC without any ac or dc control power. Blackrun of RCIC refers to the long-term operation of RCIC without electricity, once it has been started. This typically involves using a portable generator to supply power to indications such as reactor pressure vessel (RPV) level to allow the operator to manually adjust RCIC flow to prevent RPV overfill and flooding of the RCIC turbine.

** If the RCIC system is successfully controlled (i.e., successful blackstart and blackrun) then both mitigated Short-term SBO scenarios would be functionally similar to the mitigated Long-term SBO (i.e., no core damage). This was qualitatively determined based on the timing and equipment availabilities from the other SBO analyses.

*** Estimated risks below 1 x 10^{-7} per reactor-year should be viewed with caution because of the potential impact of events not studied in the analyses and the inherent uncertainty in very small calculated numbers.

Table 7 Surry Results for Scenarios Assuming LNT Dose-Response Model

Scenario	Core damage frequency (CDF) (per reactor-year)	Mitigated		Unmitigated	
		Conditional scenario-specific probability of latent cancer fatality for an individual located within 10 miles	Scenario-specific risk (CDF x Conditional) of latent cancer fatality for an individual located within 10 miles (per reactor-year)	Conditional scenario-specific probability of latent cancer fatality for an individual located within 10 miles	Scenario-specific risk (CDF x Conditional) of latent cancer fatality for an individual located within 10 miles (per reactor-year)
Long-term SBO	2×10^{-5}	No Core Damage		5×10^{-5}	~ 7×10^{-10} ***
Short-term SBO	2×10^{-6}	No Containment Failure *		9×10^{-5}	~ 1×10^{-10} ***
Short-term SBO with TISGTR	4×10^{-7}	3×10^{-4} **	~ 1×10^{-10} ***	3×10^{-4}	~ 1×10^{-10} ***
Interfacing systems LOCA	3×10^{-8}	No Core Damage		3×10^{-4}	~ 9×10^{-12} ***

* Accident progression calculations showed that source terms in the mitigated case are smaller than in the unmitigated case. Offsite consequence calculations were not run, since the containment fails at about 66 hours. A review of available resources and emergency plans shows that adequate mitigation measures could be brought onsite within 24 hours and connected and functioning within 48 hours. Therefore 66 hours would allow ample time for mitigation through measures transported from offsite.

** Containment failure is delayed by about 46 hours in the mitigated case relative to the unmitigated case. Rounding to one significant figure shows conditional LCF probabilities of $3x10^{-4}$ for both mitigated and unmitigated cases, however the original values were $2.8x10^{-4}$ for the mitigated case and $3.2x10^{-4}$ for the unmitigated case.

*** Estimated risks below 1×10^{-7} per reactor year should be viewed with caution because of the potential impact of events not studied in the analyses and the inherent uncertainty in very small calculated numbers.

To provide perspective on alternative low-dose health effect modeling, the SOARCA project has also developed LCF risk estimates assuming non-LNT models, which are based on the premise that below a certain dose, cancer risks cannot be reliably quantified, or are nonexistent. Dose truncation values used for SOARCA included 620 mrem/year (representative background radiation including average annual medical exposures), and 5 rem/year with a 10-rem lifetime cap (based on the Health Physics Society's position that there is a dose below which, because of uncertainties, a quantified risk should not be assigned). Table **8** and Table 9 show the results of sensitivity calculations for dose truncation values compared with LNT results. Using these truncation values makes the already small scenario-specific LCF risk calculations even smaller, in some cases, by orders of magnitude.

For Surry scenarios except ISLOCA, the background results in Table 9 differ from the HPS results, because the background truncation value clearly falls below the plant-specific population return criterion of 4 rem over 5 years, which is intended to represent EPA's (adopted in Virginia) criterion of 2 rem in the first year and 0.5 rem/year in subsequent years; however, the HPS truncation value does not. The ISLOCA results are the same to one significant digit within a radius of 10 miles for both truncation values, because most of the emergency phase doses exceed both of these criteria, while, on the other hand, long-term doses make an insignificant contribution to the overall doses. The results in Table 8 and Table 9 assume that the probability of 10 CFR 50.54(hh) mitigation is zero.

SOARCA analyses included predictions of individual scenario-specific LCF risk for several distance intervals, including 0 to 10 miles and 0 to 50 miles. The analysis indicated that individual LCF risk estimates generally decrease with increasing distance, in large part because of plume dispersion and fission product deposition closer to the site.

Table 8 Peach Bottom Results for Scenarios without Successful Mitigation for LNT and Alternative Dose-Response Models

Scenario	Scenario-specific risk of latent cancer fatality for an individual located within 10 miles (per reactor-year)		
	Linear No-Threshold	Background	Health Physics Society
Long-term SBO	3×10^{-10}	2×10^{-12}	1×10^{-12}
Short-term SBO	6×10^{-11}	4×10^{-12}	4×10^{-12}
Short-term SBO with RCIC Blackstart	2×10^{-11}	2×10^{-13}	9×10^{-14}

Table 9 Surry Results for Scenarios without Successful Mitigation for LNT and Alternative Dose-Response Models

Scenario	Scenario-specific risk of latent cancer fatality for an individual located within 10 miles (per reactor-year)		
	Linear No-Threshold	Background	Health Physics Society
Long-term SBO	7×10^{-10}	6×10^{-12}	2×10^{-14}
Short-term SBO	1×10^{-10}	5×10^{-12}	2×10^{-14}
Thermally induced steam generator tube rupture	$1x10^{-10}$	3×10^{-11}	5×10^{-12}
Interfacing systems LOCA	9×10^{-12}	2×10^{-12}	1×10^{-12}

Because the SBO scenarios were seismically induced, the study added analyses to evaluate the potential impact of the seismic event on the evacuation. Although road network infrastructure may be damaged during an earthquake, resulting in reduced evacuation speeds, other effects such as wider deployment of emergency responders and a larger shadow evacuation may improve evacuation timing. The analyses for both Surry and Peach Bottom indicated changes to the evacuation resulting from the earthquake would change the LCF risk by less than 10 percent and may actually cause the consequences from radionuclide release to decrease as in the case of the Peach Bottom plant, because the population is on alert after the earthquake.

6.4 Comparison to NUREG/CR-2239 (the 1982 Siting Study SST1 case)

The SOARCA offsite early fatality risk calculations are dramatically smaller than reported in NUREG/CR-2239 [1]. This Siting Study predicted 92 early fatalities for Peach Bottom and 45 early fatalities for Surry for the SST1 source term. In contrast, SOARCA predicted that the early fatality risk was essentially zero for both sites.

For LCF results, the exact basis for NUREG/CR-2239 estimates could not be recovered, but literature searches and sensitivity analyses with MACCS2 suggested that these estimates are for the population within 500 miles of the site. Moreover, an attempt to reproduce the results of NUREG/CR-2239 led to agreement within about a factor of 2. Given the uncertainty in the basis for these results, the SOARCA study performed an additional set of calculations to enable the current, state-of-the-art results to be compared with the 1982 Siting Study. For this set of calculations, the most severe source terms predicted by the SOARCA analyses (see Figure and Figure) were replaced by the largest source term from the Siting Study—the SST1 source term. No other modeling or parameter changes were made, including the timing of public evacuation. Thus, this comparison does not attempt to replicate the Siting Study; it simply evaluates the largest source term, SST1, from that study and compares the results with those from the current work.

Table 10 and Table 11 summarize the comparison to the Siting Study source term results for the Peach Bottom and Surry sites, respectively, assuming an LNT dose-response function. Although the SST1 source term is identical in both comparisons, the scenario-specific LCF probabilities associated with this source term shown in the tables are different because of the difference in evacuation modeling and other offsite consequence parameters for the two sites.

Table 10 Conditional (i.e., assuming accident occurs), Mean, LNT, Scenario-Specific Probabilities of LCF for People within the Specified Radii of the Peach Bottom Site

Radius of Circular Area (mi)	1982 Siting Study SST1	SOARCA Unmitigated STSBO
10	3.3×10^{-3}	2.1×10^{-4}
20	1.8×10^{-3}	5.7×10^{-4}
50	4.6×10^{-4}	1.9×10^{-4}

Table 11 Conditional, Mean, LNT, Scenario-Specific Probabilities of LCF for People within the Specified Radii of the Surry Site

Radius of Circular Area (mi)	1982 Siting Study SST1	SOARCA Unmitigated ISLOCA	SOARCA Unmitigated STSBO with TISGTR
10	1.0×10^{-2}	3.0×10^{-4}	3.2×10^{-4}
20	5.1×10^{-3}	3.4×10^{-4}	1.9×10^{-4}
50	1.5×10^{-3}	1.6×10^{-4}	6.5×10^{-5}

For the 0-10 mile radius, the area associated with the NRC Safety Goal for latent cancers, the scenario-specific probabilities of latent cancer fatality calculated for SOARCA are substantially smaller than predicted in the 1982 Siting Study for SST1. Considering both the Peach Bottom and Surry comparisons, this difference diminishes with increasing radius, falling from a factor of 33 within 10 miles to about a factor of about 2.4 within a 50-mile radius and beyond. Figure 14 provides additional comparisons of SOARCA results for both mitigated and unmitigated scenarios to 1982 Siting Study SST1 results for people within 10 miles of the plant.

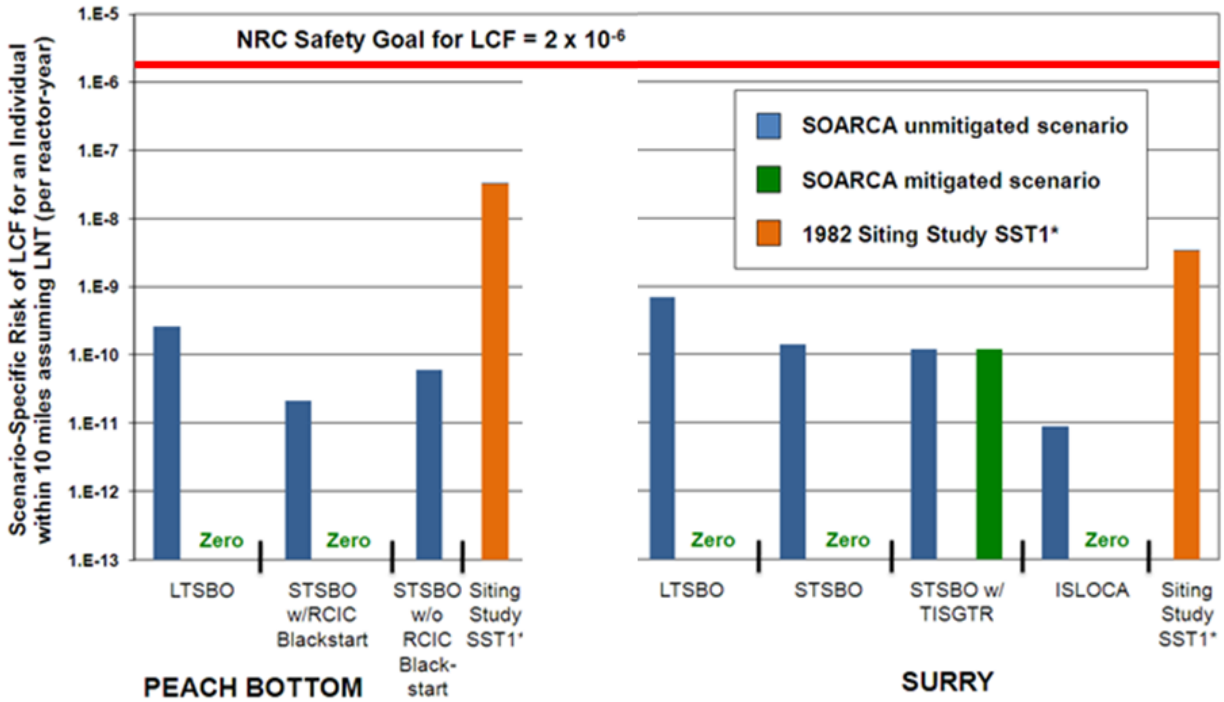


Figure 14 Comparison of individual LCF risk results for SOARCA mitigated and unmitigated scenarios to 1982 Siting Study SST1 Extrapolations and NRC Safety Goal

(Note: plotted on logarithmic scale)

6.5 Sensitivity Analyses on the Size of the Evacuation Zone

As discussed earlier in Section 5.3.2, SOARCA included sensitivities to analyze the effects of increased evacuation sizes on scenario-specific latent cancer fatality risks for people within different distances of each plant. These sensitivities were performed for distances of 16 miles and 20 miles from the plant and compared to the base case of a 10 mile radius evacuation. Selected results are presented below in Table 12 and Figures 15 and 16; however more details are provided in NUREG/CR-7110, Volumes 1 and 2. The Peach Bottom unmitigated STSBO scenario without RCIC blackstart and the Surry unmitigated ISLOCA scenario were chosen for these sensitivities because they result in the largest releases of radioactive materials of all scenarios analyzed for each plant. The Peach Bottom unmitigated STSBO scenario without RCIC blackstart releases about 12% of the core inventory of I-131 and 2% of the Cs-137 to the environment and begins at 8 hours. The Surry unmitigated ISLOCA scenario releases about 16% of the core inventory of I-131 and 2% of the Cs-137 and begins at about 13 hours.

Table 12 Effect of Size of Evacuation Zone on Mean, Individual, LNT, Scenario-Specific LCF Risk for People within the Specified Radii of the Plant for the Peach Bottom Unmitigated STSBO without RCIC Blackstart and the Surry Unmitigated ISLOCA

Radius of Circular Area (mi)	Peach Bottom Unmitigated STSBO without RCIC Blackstart		Surry Unmitigated ISLOCA	
	Base Case 10-mile Evacuation	Sensitivity 20-mile Evacuation	Base Case 10-mile Evacuation	Sensitivity 20-mile Evacuation
10	6×10^{-11}	2×10^{-10}	9×10^{-12}	1×10^{-11}
20	2×10^{-10}	7×10^{-11}	1×10^{-11}	8×10^{-12}
30	1×10^{-10}	8×10^{-11}	8×10^{-12}	7×10^{-12}
40	7×10^{-11}	6×10^{-11}	6×10^{-12}	5×10^{-12}
50	6×10^{-11}	5×10^{-11}	5×10^{-12}	4×10^{-12}

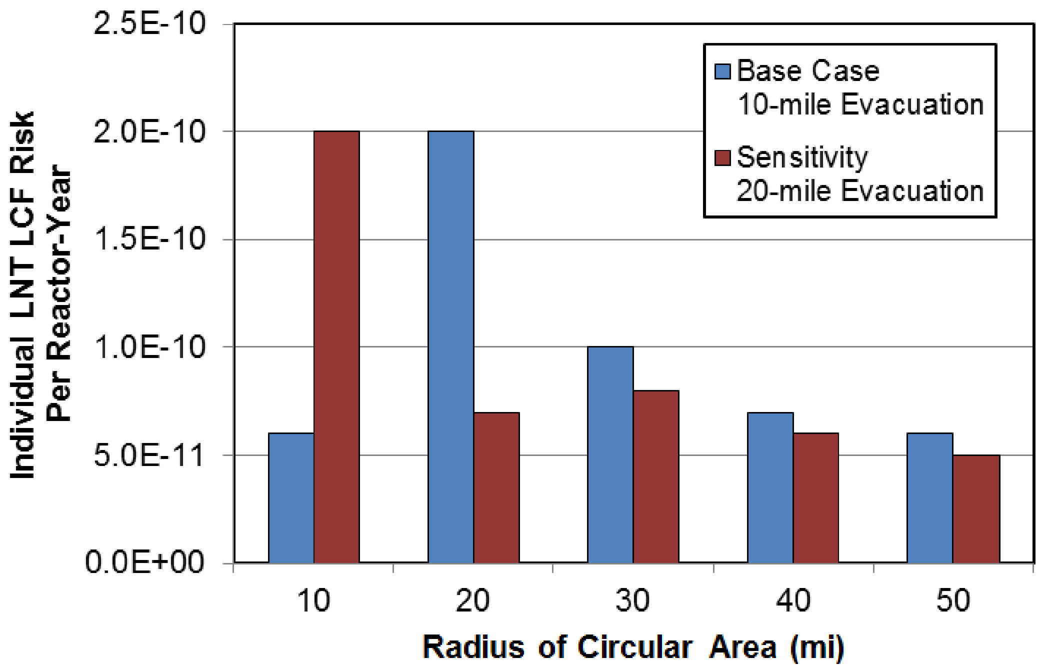


Figure 15 Mean, individual, LNT, scenario-specific LCF risk for the Peach Bottom unmitigated STSBO scenario without RCIC blackstart for people within a circular area of specified radius from the plant.

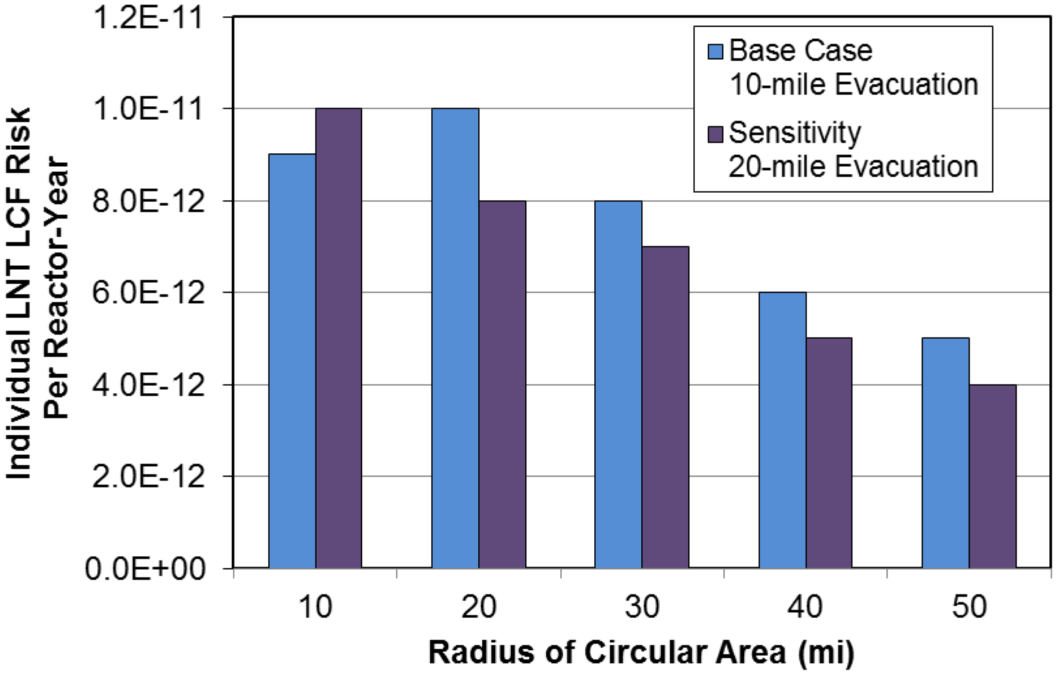


Figure 16 Mean, individual, LNT, scenario-specific LCF risk for the Surry unmitigated ISLOCA scenario for people within a circular area of specified radius from the plant.

Analysis of the evacuation size sensitivities for both Peach Bottom and Surry shows that for the base case, LCF risk is slightly higher for people within a 20 mile radius of the plant compared to people within a 10 mile radius of the plant. This result is likely due to SOARCA's modeling assumptions rather than physical reality. SOARCA models a shadow evacuation, which is the voluntary (self-initiated) evacuation of members of the public from areas that are not under official evacuation orders, yet it does not model an evacuation that (for which there is no preplanning) would be ordered by officials for certain areas outside the EPZ. Therefore, the slight increase in LCF risk in the base case for people within a 20 mile radius of the plant relative to people within a 10 mile radius is not considered a meaningful insight.

Analysis of Figure 15 and 16 also shows that expanding the evacuation size from a 10 mile radius to a 20 mile radius results in increased LCF risk for people in the 0-10 mile area. SOARCA analyses show that an evacuation beyond the area closest to the plant will delay those most at risk, i.e., closest to the plant. The increased risk to the population within 10 miles of the plant is due to slower evacuation speeds because of additional traffic congestion and delays that result from evacuation of a larger population.

Further, SOARCA's evacuation size sensitivities show that an evacuation out to 20 miles from the plant results in decreased risk for the population relative to the base case 10 mile evacuation. For the Peach Bottom unmitigated STSBO without RCIC blackstart, the scenario-specific LCF risk falls from 2 $\times 10^{-10}$ per reactor-year to about 7 $\times 10^{-11}$, a factor of about 3, while the risk reduction is much smaller for the Surry unmitigated ISLOCA scenario. The decrease in average LCF risk out to 20 miles is seen as within the bounds of modeling assumptions and not indicative of a measurable benefit. For the 20 mile evacuation, SOARCA did not model a shadow evacuation; however if this was included, it would likely delay the evacuation of the people within 20 miles and increase the scenario-specific LCF risk. This likely leads to an underestimation in scenario-specific LCF risk for the people within 20 miles during the 20 mile evacuation.

Overall, the increases and reductions to scenario-specific LCF risk shown in Table 12 and Figure 15 and 16 are extremely small on an absolute scale. The LCF risks calculated for SOARCA's base case and 20-mile evacuation sensitivity are all millions of times smaller than the average annual risk of cancer death for an individual in the United States.

6.6 Conclusions

The results of the SOARCA project represent a major change in the staff's perception of severe reactor accidents and their consequences. Specific conclusions of the project are as follows:

The SOARCA results demonstrate the potential benefits of employing 10 CFR 50.54(hh) mitigation enhancements for the scenarios analyzed. When successful mitigation is assumed, the MELCOR results indicate no core damage for all scenarios except the Surry STSBO and its TISGTR variant. For the Surry STSBO with mitigation, the core is damaged; however, containment failure is delayed by an additional 41 hours compared to the unmitigated case. The mitigation measures (i.e., containment sprays) are effective in knocking down the airborne aerosols. For the Surry STSBO with TISGTR with mitigation, the core is damaged and containment failure is delayed by an additional 46 hours compared to the unmitigated case. This is a bypass scenario, and therefore the release to the environment begins at the same time as in the unmitigated case. For both the mitigated and unmitigated cases, the individual scenario-specific LCF risk for the EPZ was small, approximately 1×10^{-10} per reactor-year, assuming an LNT dose-response model.

When the scenarios were assumed to proceed unmitigated (i.e., neither 10 CFR 50.54(hh) implementation nor other key operator actions that would prevent core damage), MELCOR analyses indicated that the accidents progress more slowly and with smaller releases than the 1982 Siting Study SST1. Whereas the 1982 Siting Study SST1 case results in a large early release at 1.5 hours, the SOARCA analyses show no large early releases for the scenarios analyzed.

The individual early fatality risk from SOARCA scenarios is essentially zero. Individual LCF risk from the selected specific, important scenarios is thousands of times lower than the NRC Safety Goal and millions of times lower than the general cancer fatality risk in the United States from all causes, even assuming the LNT dose-response model. Using a dose-response model that truncates annual doses below normal background levels (including medical exposures) results in a further reduction to the LCF risk (by a factor of 100 for smaller releases and a factor of 3 for larger releases). LCF risk calculations are generally dominated by long-term exposure to small annual doses (about 500 mrem per year) corresponding to evacuees returning to their homes after the accident and being exposed to residual radiation over a long period of time.

SOARCA results indicate that bypass events (e.g., Surry ISLOCA) do not pose a higher scenario-specific latent cancer fatality risk than non-bypass events (e.g., Surry SBO). While consequences are greater when the bypass scenario happens, this is offset by the scenario being less likely to happen. SOARCA reinforces the importance of external events relative to internal events and the need to continue ongoing work related to external events risk assessment.

The SOARCA analyses show that emergency response programs, implemented as planned and practiced, reduce the scenario-specific risk of health consequences among the public during a severe reactor accident. Sensitivity analyses of seismic impacts on site-specific emergency response (e.g., loss of bridges, traffic signals, and delayed notification) at Peach Bottom and Surry do not significantly affect LCF risk.

SOARCA results, while specific to Peach Bottom and Surry, may be generally applicable for plants with similar designs. However, additional work is needed to confirm this, since differences exist in plant-specific designs, procedures, and emergency response.

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Appendix A to NUREG-1935: State-of-the-Art Reactor Consequence Analyses Project and the Fukushima Daiichi Accident

Objective

The State-of-the-Art Reactor Consequence Analyses (SOARCA) study was nearly at the end of its peer review when the Fukushima Daiichi accident occurred on March 11, 2011. Following the accident, the U.S. Department of Energy (DOE) and the U.S. Nuclear Regulatory Commission (NRC) began a cooperative effort to use the MELCOR code for a forensic analysis of event progression to develop a more detailed understanding of the accident. This cooperative effort is ongoing.

Based on limited information currently available, the Fukushima accident has many similarities and differences with some of the Peach Bottom sequences analyzed in SOARCA. The objective of this appendix is to compare and contrast the Fukushima accident and the SOARCA study for the following topics: (1) operation of the reactor core isolation cooling (RCIC) system, (2) hydrogen release and combustion, (3) 48-hour truncation of releases in SOARCA, (4) multiunit risk, and (5) spent fuel pool (SFP) risk. It must be emphasized that there are significant gaps in information and uncertainties about what actually occurred in the Fukushima reactors. These uncertainties do not allow firm conclusions on comparisons with SOARCA results. It is expected to take a number of years for the Japanese organizations involved to be able to access the containments and fully evaluate the conditions of the nuclear fuel and other equipment to allow a more complete understanding of the events.

Background

The Great East Japan Earthquake, which rated a magnitude 9.0 on the moment magnitude scale (M_w), occurred northeast of Tokyo off the east coast of Honshu Island. This earthquake resulted in the automatic shutdown of the Fukushima Daiichi reactors. The earthquake precipitated a tsunami that exceeded 14 meters (45 feet) in height at the Fukushima Daiichi site. The earthquake and subsequent tsunami produced widespread devastation across northeastern Japan, resulting in approximately 25,000 people dead or missing, displacing many tens of thousands of people, and significantly affecting the infrastructure and industry in the northeastern coastal areas of Japan.

On March 11, 2011, Fukushima Daiichi Units 1, 2, and 3 were in operation, and Units 4, 5, and 6 were shut down for routine refueling and maintenance activities; the Unit 4 reactor fuel was offloaded to the Unit 4 SFP. The description of events below is based on our current understanding of the accident, which, as previously stated, is subject to significant uncertainty.

As a result of the earthquake, all of the operating units appeared to experience a normal reactor trip within the capability of the safety design of the plants. The three operating units (Units 1, 2, and 3) automatically shut down, inserting all control rods into the respective reactors. Also, as a result of the earthquake, offsite power was lost to the entire facility. The emergency diesel generators started at all six units providing alternating current (ac) electrical power to critical systems, and the facility response to the seismic event appears to have been normal.

Approximately 40 minutes after the earthquake and shutdown of the operating units, a large tsunami wave inundated the site, followed by multiple additional waves. The estimated height of the tsunami exceeded the height for which site protection features against tsunamis were designed by approximately 8 meters (27 feet). The tsunami resulted in extensive damage to site facilities and a complete loss of ac electrical power at Units 1 through 5 (i.e., a station blackout (SBO)). Unit 6 retained the function of one of the diesel generators.

Without ac power, the plants relied on batteries and turbine-driven and diesel-driven pumps for reactor core cooling (it should be noted that immediately after the tsunami, Units 1 and 2 were without 125 volt dc power too). The operators took actions to maintain core cooling functions well beyond the normal capacity of the station batteries. However, without sufficient offsite assistance, which appears to have been hampered by the devastation in the area, among other factors, Units 1 through 3 eventually lost the ability to further extend cooling of the reactor cores. This ultimately resulted in significant damage to the reactor cores in these units, the extent of which is still the subject of evaluation.

At varying points in time after the tsunami, Units 1, 3, and 4 experienced explosions, further damaging the facilities and containment and reactor buildings. The Unit 1 and 3 explosions were apparently caused by the buildup of hydrogen gas within containment produced during fuel damage in the reactor and subsequent movement of that hydrogen gas from the drywell into the reactor building. The explosion that occurred in Unit 4 may have involved hydrogen that was transported through a ventilation system connected to Unit 3.

As information about the damage to plant safety functions was gathered over the weeks and months following these events, many similarities became apparent between the calculated damage progression in the boiling-water reactor (BWR) SBO accident scenarios in the SOARCA analyses and the progression of events at Fukushima. These similarities include the following:

- the sequence and timing of events that followed the loss of core cooling, including the onset of core damage and fission product release from fuel,

- challenges to containment integrity that accompanied the loss of decay heat removal and the accumulation of hydrogen generated during in-vessel damage to reactor fuel, and

- the destructive effects of hydrogen combustion in the reactor building.

As noted in the discussion of hydrogen combustion below, the SOARCA analyses and the Fukushima events appear to have released hydrogen to the reactor building by different mechanisms. But in both cases, the end result was structural failure of the building and radionuclide release to the environment caused by energetic combustion. Similarities were also observed in characteristics of radionuclide release to the environment in the SOARCA calculations and early measurements of activity in the areas surrounding the Fukushima site.

Some notable differences in the events that unfolded at Fukushima and the BWR long-term SBO (LTSBO) scenario studied in the SOARCA project were also readily apparent. These differences led the NRC staff to take a closer look at the models used and assumptions made in the LTSBO analyses. The NRC's SOARCA team qualitatively compared the results from the SOARCA analyses to the preliminary events and information available at this early stage in the evaluation of the Fukushima Daiichi accident, and the results are discussed below.

The information used to evaluate these topics was gleaned from a variety of sources. Most important among these are the following:

- "Report of the Japanese Government to the IAEA [International Atomic Energy Agency] Ministerial Conference on Nuclear Safety," June 2011 [[1]]

- "Special Report on the Nuclear Accident at the Fukushima Daiichi Nuclear Power Station," Institute of Nuclear Power Operations (INPO), November 2011 [7]

Shortly after the accident, the NRC established a task force to conduct a methodical and systematic review of the agency's processes and regulations to determine whether it should make additional improvements to its regulatory system. The task force report, "Recommendations for Enhancing Reactor Safety in the 21st Century" [[2]], found, among other things, that prolonged SBO and multiunit events present challenges to emergency response. The task force report presented a number of recommendations that address physical, administrative, and regulatory enhancements to further reduce the risk of similar challenges occurring among the U.S. fleet of nuclear power plants.

Operation of the Reactor Core Isolation Cooling System

According to the "Report of Japanese Government to IAEA Ministerial Conference on Nuclear Safety" [[1]], and the Institute of Nuclear Power Operations (INPO) "Special Report on the Nuclear Accident at the Fukushima Daiichi Nuclear Power Station" [[7]], the RCIC system was used to maintain coolant injection to the reactor pressure vessel (RPV) for approximately 70 hours in Unit 2 and for 21 hours in Unit 3[1]. The SOARCA study performed MELCOR analyses for the Peach Bottom station blackout scenarios shown in the following table.

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1 When the RCIC system tripped in Unit 3, a separate steam-driven high-pressure coolant injection system (HPCI) automatically started and ran for approximately 2 hours. Unit 1 has a different system (isolation condenser) and does not have the RCIC system.

	Start of RCIC operation (hours)	End of RCIC operation (hours)	Duration of RCIC operation (hours)
Fukushima Unit 2	0	70	70
Fukushima Unit 3	0	21	21
Peach Bottom unmitigated LTSBO	0	5	5
Peach Bottom unmitigated STSBO with RCIC blackstart	1	3	2
Peach Bottom unmitigated STSBO without RCIC blackstart	Not modeled	Not modeled	N/A

Note: The MELCOR analysis was truncated at 48 hours, as discussed below.

The operators at Fukushima Units 2 and 3 were able to successfully operate their RCIC systems to maintain water inventory within the core for a period of time that greatly exceeded the operating period assumed in the SOARCA calculations of the unmitigated LTSBO scenario.

In the SOARCA analysis of the unmitigated LTSBO scenario, the RCIC system operates for 4 hours while direct current (dc) power is available and an additional 1.2 hours after station batteries are exhausted[2] (i.e., RCIC blackrun). Measurement of the reactor water level would be lost at 4 hours, when station batteries that provide dc power to critical plant instrumentation are exhausted. This condition led to the assumption that manual operation of RCIC would maintain a constant (throttled) flow rate with the goal of maintaining the reactor coolant water level. However, as discussed in the detailed description of the MELCOR analysis of the unmitigated LTSBO scenario, sustained (constant flow) operation of RCIC after the loss of dc power would lead to an increase in the RPV water level and steamline flooding in approximately 1.2 hours. Flooding the main steamline would result in flooding the RCIC turbine, disabling the system. Overfilling the RPV is a consequence of the imbalance between the termination of coolant losses through safety relief valves (which reclose on a loss of dc power) and continued coolant addition by the RCIC system.

Flooding of the RCIC turbine by reactor vessel overfill does not appear to have occurred at Fukushima, and the operators successfully ran the system for an extended period of time in Units 2 and 3. One reason for the extended length of RCIC operation at Fukushima, in comparison to the unmitigated LTSBO timeline in the SOARCA analysis, is the station batteries at Fukushima were designed to provide dc power for a longer period of time than the batteries at Peach Bottom (8 hours for Fukushima versus 2 hours). At both plants, the actual duration of dc power would be longer than the design basis because of margins incorporated into the system design, as well as manual actions that can be taken to shed nonessential loads on the dc emergency bus. The maximum length of time that dc power was available at Fukushima appears to have been considerably longer than the maximum battery duration considered in the SOARCA analysis for Peach Bottom, even when load shedding is taken into account.

A second reason for the difference in RCIC operating time is that manual actions taken at Fukushima to manage RCIC operation after the loss of dc power appear to have differed from

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2 RCIC is a steam-driven coolant injection system that does not require ac power for it to start or operate as a coolant injection system. Steam flow to the RCIC turbine can be remotely controlled with dc power from the station batteries. However, manual operation using valve handwheels (i.e., blackstart, blackrun) is also possible.

those assumed in the unmitigated LTSBO scenario. The precise actions taken by TEPCO operations personnel to control RCIC flow in Units 2 and 3 are not known. However, it is clear that RCIC flow at Fukushima Units 2 and 3 was regulated at values that prevented overfill of the RPV and flooding of the main steamlines. The INPO report [7] indicates portable electric generators were installed in the control rooms, and power was restored to critical plant instrumentation, including RPV water-level measurement. This action facilitated control of RCIC flow to maintain the RPV level at desired values. The SOARCA analysis of the unmitigated LTSBO sequence did not credit staging and alignment of portable electric generators (i.e., equipment required in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.54(hh)). The 10 CFR 50.54(hh) equipment at Peach Bottom also includes portable coolant pumps, which, according to the SOARCA analyses, would satisfactorily maintain core cooling and avert core damage if aligned and operated successfully. Therefore, differences in operator actions to manage RCIC flow after the loss of dc power versus those assumed for SOARCA, as well as differences in the availability of portable mitigation equipment, contributed to differences in the timeline of events at Fukushima versus the timeline calculated for the unmitigated LTSBO scenario.

Finally, the RCIC system in Fukushima Units 2 and 3 appears to have run for many hours under conditions that exceed established operating limits for the turbine-driven pumps. The reason that the RCIC pumps eventually stopped running is not known. However, preliminary DOE/NRC forensic analysis of event progression at Units 2 and 3 indicate the torus water temperature in both units exceeded values that would have challenged pump operation because of loss of adequate net positive suction head, vibration and mechanical damage from pump cavitation, or overheating of pump bearings caused by inadequate cooling. The SOARCA models did not anticipate nor incorporate sustained endurance of the RCIC system (well beyond design limits).

These differences in the factors contributing to the duration of RCIC operation at Fukushima and the SOARCA analyses result in two differences in the observed chronology of events that follow the eventual loss of coolant injection. First is the difference in the times at which core damage and fission product release to the environment begin. These events were predicted to begin at 20 hours in the SOARCA unmitigated LTSBO analysis, but they began at Fukushima Units 2 and 3 on the third and second day of the accident, respectively. Second, sustained operation of the RCIC system at Fukushima resulted in a larger cumulative transport of heat from fission product decay (in the form of steam) from the RPV to the suppression pool in the containment (torus). Suppression pool temperatures at the time the RCIC pumps ceased operating in Fukushima Units 2 and 3 were, therefore, much higher than the calculated pool temperature in the SOARCA analysis of the unmitigated LTSBO scenario. Increases in suppression pool temperature result in additional evaporation of water from the pool to the containment atmosphere; this in turn results in an increase in containment pressure. Therefore, containment pressure in the Fukushima reactors at the time core damage began was higher than the pressure calculated in the Peach Bottom LTSBO scenario. When hydrogen, generated by oxidation of Zircaloy cladding in the core, was released to the containment atmosphere, containment pressure increased further. The combination of a high base pressure from long-term evaporation of steam and accumulation of noncondensible hydrogen gas in the Fukushima containments likely resulted in pressures that were sufficiently high to induce leakage through the drywell head flange while in-vessel core damage was underway. Release of hydrogen to the reactor building through the drywell head flange likely led to the destruction of the Fukushima reactor buildings by hydrogen combustion. In contrast, the shorter duration of RCIC operation in the unmitigated LTSBO scenario resulted in less heating of the suppression pool, less evaporation of water to the containment atmosphere, and a lower base pressure in containment at the time core damage and hydrogen generation began.

The extended period of core cooling by sustained RCIC operation at Fukushima affected more than the timeline for core damage and containment pressure at the time core damage began. The mechanisms for hydrogen (and fission product) leakage out of containment into the reactor building were affected by differences in containment thermodynamic conditions, which were influenced by the operation of the steam-driven RCIC system.

Hydrogen Release and Combustion

The physical damage to Fukushima reactor buildings will perhaps be the most enduring visible image of plant damage initiated by the earthquake and tsunami in Japan in March 2011. The apparent cause was combustion of hydrogen that was generated by high-temperature oxidation of fuel cladding. Extensive cladding oxidation and core material melting is believed to have occurred in Fukushima Units 1, 2, and 3, although the timelines for core damage differed in each unit because of differences in equipment and operator response.

At Fukushima Units 1 and 3, hydrogen generated from the oxidation of fuel cladding in the core was likely transported from the RPV to the containment (drywell and wetwell) through an open or cycling safety relief valve. Hydrogen is predicted to be released to the containment by the same pathway in the unmitigated SOARCA scenarios. The precise pathway by which hydrogen was released from the containment to the reactor building at Fukushima is uncertain. The Japanese Report to the IAEA [[1]] suggests the pathway was leakage through the drywell head flange, which is normally sealed by a pair of O-ring seals. High internal pressures developed within the drywell as a consequence of the failure of engineered systems for containment heat removal from loss of ac power from various causes and the accumulation of noncondensible hydrogen. The resulting mechanical loads transmitted to the drywell head and closure bolts are believed to have resulted in leakage past the closure seals, through the head flange to the upper portion of the reactor building. The Japanese government report [[1]] suggests this leakage pathway developed in all three units in which core damage occurred (Units 1, 2, and 3).

Leakage across the drywell head flange is modeled in the SOARCA scenarios. Opening criteria for this leak pathway are based on NRC calculations of the internal pressure required to cancel the compressive force on the closure head flange created by the torque applied to the head bolts. Only one of the SOARCA calculations (unmitigated LTSBO) resulted in a drywell pressure sufficiently high to open this release pathway before drywell liner melt-through. In this case, the liner melt-through occurs shortly after head flange leakage begins. As a result, in the SOARCA scenarios, significant hydrogen release to the reactor building occurs only after mechanical failure of the containment pressure boundary, which in the SOARCA calculations results from molten debris failing the drywell liner after RPV lower head failure (i.e., drywell liner melt-through). Containment failure by this mechanism is not believed to have occurred in any of the units at Fukushima. However, the Japanese government report suggests the possibility of some amount of molten debris being released from the RPV lower head to the drywell floor within the reactor pedestal in Units 2 and 3. Additionally, TEPCO has announced that a recent analysis of Unit 1 (accident simulations performed with the MAAP computer code) suggest the bulk of the fuel was released into the drywell through RPV lower head failure [1]. TEPCO concluded from this analysis that fuel released to the drywell floor eroded approximately 70 centimeters (2.3 feet) of concrete on the drywell floor within the reactor pedestal, but the fuel did not move laterally across the drywell floor.

The BWR MELCOR model used in the SOARCA calculations ignites and burns hydrogen in regions of the reactor building where local concentrations satisfy assumed flammability criteria (see NUREG/CR-7110, Volume 1). The pressure generated by hydrogen combustion within the reactor building results in opening the blowout panels in the walls of the refueling floor. In the station blackout scenarios examined in the SOARCA calculations, the combustion pressure is sufficiently high to also fail (open) many doorways within the building (e.g., into and out of the stairwells) and the large railroad access doorways to the environment at grade level. Structural failure of the steel roof of the reactor building can also occur, if these pathways are insufficient to relieve the internal pressure generated by hydrogen combustion. Roof failure was calculated to have occurred in the SOARCA short-term SBO (STSBO) scenario but not in the LTSBO scenario.

Generation of hydrogen from oxidation of fuel stored in the SFP of Fukushima Unit 4, which was shut down for maintenance at the time of the accident, is not believed to have occurred in significant quantities. However, the Unit 4 reactor building was severely damaged, apparently by the combustion of hydrogen that leaked into the building. It has been proposed that the source of hydrogen in the Unit 4 reactor building was hydrogen that flowed through piping that connects the standby gas treatment system (SGTS) in Unit 3 to a parallel system in Unit 4. Hydrogen may have entered the SGTS system in Unit 3 when containment venting was performed, because the containment vent system at Fukushima allows the operator to direct gases through the SGTS for filtration before being released to the stack[3]. Although venting through a similar pathway is possible at Peach Bottom (e.g., opening the containment ventilation system, which is connected to the SGTS), a different release pathway for containment venting is simulated in the SOARCA calculations for Peach Bottom[4]. Virtually all BWR Mark I containments in the United States have a hardened vent that bypasses the normal containment ventilation system and associated SGTS and vents the containment atmosphere directly to the environment. This pathway bypasses the SGTS filters and is comprised of rigid piping rather than the thin metallic ductwork common to building ventilation systems. As a result, hydrogen gas would be discharged to the environment, rather than leaking to the reactor building through leaks or ruptures in ventilation duct work, which would not likely survive the internal pressure

3 The precise configuration of the containment vent pathway used at Fukushima is not clear in terms of its discharge location relative to the SGTS filters. The INPO report describes a configuration that bypasses SGTS. However, as noted earlier, SGTS ductwork is believed to be the path by which hydrogen flowed from Unit 3 into Unit 4.

4 Selection of a containment venting pathway is a proceduralized action at Peach Bottom and includes an assessment of potential adverse characteristics of each pathway. For example, venting through the drywell (or wetwell) ventilation system to SGTS could adversely affect the environment in the reactor building if relatively weak ventilation ductwork were to fail because of high internal pressure, releasing hydrogen and radioactivity into the reactor building. This could cause accessibility issues for other operator actions.

anticipated if containment venting were to become necessary. Therefore, hydrogen leakage to the reactor building would not occur as a result of containment venting if the hardened vent is used, as assumed in the SOARCA models. While hardened vents that allow the operator to bypass the SGTS filters were installed at Fukushima between 1999 and 2001, it is unclear whether they were used during the March 2011 events.

48-Hour Truncation of Releases in SOARCA

The 48-hour truncation time for SOARCA was based on the many resources available at the State, regional, and national level that would be available to mitigate a severe reactor accident. The staff reviewed available resources and emergency plans and determined that adequate mitigation measures (at minimum, the ability to flood the reactor building) could be brought onsite within 24 hours and connected and functioning within 48 hours. The decision to truncate releases at 48 hours (72 hours for the Surry LTSBO) was made well before the Fukushima accident. Based on the assumptions made for SOARCA, the releases that would occur within 48 hours for the Peach Bottom unmitigated scenarios cease because of reactor building flooding. For Fukushima, as discussed above, the operators delayed releases beyond the SOARCA assumption, so substantial releases occurred beyond 48 hours. In addition, the operators at Fukushima were not able to flood the reactor buildings, as assumed for SOARCA.

For mitigated cases, the SOARCA analysis assumed the effectiveness of mitigation measures well within 48 hours. This assumption is considered reasonable, given the vast network of resources available in the United States. These resources include an offsite emergency operations facility, which would provide access to fleetwide emergency response personnel and equipment, including the 10 CFR 50.54(hh) mitigation measures and equipment from sister plants. These assets, as well as those from neighboring utilities and State preparedness programs, could be brought to bear on the accident if needed. In addition, SOARCA did not assume a tsunami, and such an event is considered highly unlikely at Peach Bottom and Surry. If sites were subject to tsunamis, these events could affect the availability and effectiveness of mitigation measures. In response to the recommendation of the NRC's Near Term Task Force report, SECY-11-0093, dated July 12, 2011, the NRC is currently evaluating whether changes to mitigation strategies are warranted.

Multiunit Risk

As demonstrated by the Fukushima accident, severe accidents that affect multiple reactors located at a common site are possible. Such accidents may happen following an initiating event that simultaneously challenges all reactors (e.g., earthquakes, tsunamis, loss of the electrical power grid) or following an accident in a single reactor that cascades to other reactors through interconnected electric power or cooling water systems, or inaccessibility to areas of the plant because of an ongoing radioactive release at one unit. An example of physical interactions for multiunit risk is the ad hoc installation of a temporary power cable from a mobile electric power supply to the standby liquid control pump in the Fukushima Unit 2 reactor building [[7]]. TEPCO personnel completed their work to install this equipment minutes before the explosion in the Unit 1 reactor building occurred. Debris generated by the explosion in Unit 1 damaged the temporary cables and the power supply vehicle in Unit 2, defeating earlier actions to recover the standby liquid control pump as a resource for high-pressure coolant injection.

Although beyond the scope of the SOARCA project, the NRC staff previously recognized the potential risks of multiunit accidents and has taken steps to further analyze them. As a result of the SOARCA analyses, the NRC established a generic issue to further consider the implications of multiunit accidents. Subsequently, in SECY-11-0089 [[3]], the staff proposed a site-wide Level 3 probabilistic risk assessment (PRA) that included an analysis of multiunit accidents initiated by internal and external causes during any plant operating mode. The scope of this proposed analysis includes all spent fuel stored onsite either in SFPs or in dry casks in addition to all reactors. The proposed analysis would assess the radiological consequences from multiple releases that may occur at separate times. In its staff requirements memorandum dated September 21, 2011 [[3]], the Commission directed the staff to complete the site-wide Level 3 PRA project within 4 years.

Spent Fuel Pool Risk

The SOARCA analyses did not include impacts on the spent fuel pools (SFPs) for either Peach Bottom or Surry. However, various recent risk studies, most recently NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," issued February 2001 [[4]], have shown that storage of fuel in a high-density configuration in SFPs is safe and that the risk of a significant release of radioactive materials as a result of loss of SFP cooling is expected to be less than reported in previous studies. More advanced analyses of SFPs have been conducted as part of the NRC's post-9/11 security assessments. However, these are not publicly available because of their sensitive nature. The agency has since restated its views that spent fuel is stored safely in high-density configurations in a response to SECY-08-0036, "Denial of Two Petitions for Rulemaking Concerning the Environmental Impacts of High-Density Storage of Spent Nuclear Fuel in Spent Fuel Pools (PRM [Petition for Rulemaking]-51-10 and PRM-51-12)," dated June 19, 2008 [[5]], as well as the revision to NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants—Draft Report for Comment," issued July 2009 [[6]]. Partly because of changes in the path forward of the planned Yucca Mountain geologic repository and the Fukushima accident, interest in the safety of spent fuel storage has recently increased. Therefore, the NRC has commissioned an SFP scoping study, which started in 2011, aimed at updating the estimation of radiological consequences of a severe accident on SFPs with both high-density and low-density storage configurations. The scenario considered in the study is a beyond-design-basis seismic event in the range of 0.5 to 1 g peak ground acceleration. The study involves seismic and structural analysis of the earthquake and its effects on the SFP; thermal-hydraulic and severe accident progression modeling with the MELCOR computer code; emergency preparedness and response; and, finally, offsite consequence analyses with the MACCS2 code. The plan is to document the results of the study in a publicly available report within the next year.

In the analyses presented in this report, hydrogen produced by oxidation of Zircaloy, whether produced in-vessel during core degradation, or ex-vessel by core-concrete interactions, is predicted to be burned in compartments of the reactor building as released via the failure of the drywell liner by melt-attack. These burns occur as flammability conditions are attained, mainly as hydrogen concentrations increase to the point that combustion can occur. The burns produce sufficient building over pressure to open the refueling bay blowout panels and blow open doors in the building. The explosions that were observed in the accidents at Fukushima were significantly larger than predicted in the SOARCA analyses, perhaps involving detonations where ignition might have taken place at higher concentrations than predicted in SOARCA. The damage to the Fukushima Unit 3 refueling bay was especially significant with building debris (steel and concrete) falling into the spent fuel pool, also located in the refueling bay. The debris observed in the Unit 3 spent fuel pool may well have mechanically damaged some of the fuel assemblies stored there; however, isotopic analysis of the pool water performed by TEPCO for radioactive contamination does not suggest any significant releases from the fuel rods. Moreover, the water of the spent fuel pool provides massive scrubbing capability for any released fission products such that this potential source for environmental release becomes vanishingly small. The structural damage could, on the other hand, present engineering challenges for maintaining long term cooling of the fuel stored in the pool in the days and weeks following the accident. For these reasons, SOARCA did not consider source terms from ancillary damage to the spent fuel pool from hydrogen deflagrations.

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Accident phenomena and offsite consequences of severe reactor accidents have been the subjects of considerable research over the last several decades by the U.S. Nuclear Regulatory Commission (NRC). As a consequence of this research focus, analyses of severe accidents at nuclear power reactors are more detailed, integrated, and realistic than at any time in the past. A desire to leverage this capability to address conservative aspects of previous reactor accident analyses was a major motivating factor in the genesis of the State-of-the-Art Reactor Consequence Analyses (SOARCA) project. By applying modern analysis tools and techniques, the SOARCA project developed a body of knowledge regarding the realistic outcomes of severe nuclear reactor accidents. To accomplish this objective, the SOARCA project's integrated modeling of accident progression and offsite consequences used both state-of-the-art computational analysis tools and best modeling practices drawn from the collective wisdom of the severe accident analysis community. This study has focused on providing a realistic evaluation of accident progression, source term, and offsite consequences for select scenarios for the Peach Bottom Atomic Power Station and Surry Power Station. By using the most current emergency preparedness practices and plant capabilities, as well as the best available modeling, these analyses are more realistic than past analyses. These analyses also consider mitigative measures (e.g., emergency operating procedures, severe accident management guidelines, and Title 10 to the Code of Federal Regulations (10 CFR) 50. 54(hh) measures), contributing to a more realistic evaluation.

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