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State-of-the-Art Reactor Consequence Analysis (SOARCA) Project SOARCA Methods Volume I

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ABSTRACT

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The evaluation of accident phenomena and the offsite consequences of severe reactor accidents has been the subject of considerable research by the NRC over the ~~last~~ several decades. As a consequence of this research focus, analyses of severe accidents at nuclear power reactors is more detailed, integrated and realistic than at any time in the past. ~~A desire to leverage this~~ e capability to ~~address excessively conservative~~ aspects of previous reactor accident analysis ~~efforts~~ was a major motivating factor in the genesis of the State-of-the-Art Reactor Consequence Analysis (SOARCA) project. By applying modern analysis tools and techniques, the SOARCA project seeks to provide a body of knowledge that will support an informed public understanding of the likely outcomes of severe nuclear reactor accidents. The primary objective of the SOARCA project is to provide a best estimate evaluation of the likely consequences of important severe accident events at reactor sites in the U.S. civilian nuclear power reactor fleet. To accomplish this objective the SOARCA project utilized integrated modeling of accident progression and off site consequences using both state-of-the-art computational analysis tools as well as best modeling practices drawn from the collective wisdom of the severe accident analysis community.

include new information and changes in design and operation of reactors compared with

Paperwork Reduction Act Statement

The information collections contained in this NUREG are covered by the requirements of 10 CFR Parts 50, 52, and 110, which were approved by the Office of Management and Budget, approval number 3150-0011, -0151 and -0036.

Public Protection Notification

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

Executive Summary for the Full NUREG for Peach Bottom and Surry

Background and Objective

The evaluation of accident phenomena and offsite consequences of severe reactor accidents has been the subject of considerable research by the U.S. Nuclear Regulatory Commission (NRC), the nuclear power industry, and the international nuclear energy research community. As part of an NRC initiative to assess plant response to security-related events, updated analyses of severe accident progression and offsite consequences were completed utilizing the wealth of accumulated research and incorporating more detailed, integrated, and realistic modeling than past analyses. The results of those security-related studies confirmed and quantified what was suspected but not well-quantified —namely, that some past studies of plant response and offsite consequences (for non-security events) were conservative to the point that predictions were not useful for characterizing results and guiding public policy. The subsequent misuse and misinterpretation of these estimates further suggests that communication of risk attributable to severe reactor accidents should be based on realistic estimates of the more likely outcomes.

The State-of-the-Art Reactor Consequence Analyses (SOARCA) project involves the reanalysis of severe accident consequences to develop a body of knowledge regarding the realistic outcomes of severe reactor accidents. In addition to incorporating the results of over 25 years of research into severe accident phenomenology and an understanding of the effects of radiation exposure on humans, the objective of this study is to include important plant changes (e.g., system improvements, training and emergency procedures, offsite emergency response improvements, power uprates and higher core burnup) which have been made by plant owners and are not reflected in earlier NRC assessments. The changes evaluated also include those enhancements recently made in connection with security-related events. Thus, a key component of this study was to evaluate the benefits of the recent mitigation improvements in preventing core damage events or in minimizing the offsite release should one occur. The NRC expects that the results of the SOARCA study would provide the foundation for communicating severe-accident-related aspects of nuclear safety to Federal, State, and local authorities; licensees; and the general public. This evaluation of severe accident consequences also would update the quantification of offsite consequences found in earlier NRC publications such as NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," dated December 1982, known as the Sandia Siting Study, and NUREG/CR-2723, "Estimates of the Financial Consequences of Reactor Accidents," dated September 1982, known as the Strip Report. NUREG/CR-2239 and NUREG/CR-2723 are companion documents and reference is made to both in the SOARCA documentation. For clarity, when referring to these documents, we will reference them as the Sandia Siting Study (or merely the siting study).

This report describes the analysis of two reactors, the Peach Bottom Atomic Power Station and the Surry Power Station, which served as pilot plants for the study. Peach Bottom is generally representative of a major class of U.S. operating reactors, General Electric boiling water reactor (BWR) designs that that have Mark I containments. Surry is generally representative of a second major class of U.S. operating reactors. Westinghouse pressurized water reactor (PWR) designs with large, dry containments.

Method

The approach was to utilize 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 (MACCS2 code) in a consistent manner (e.g., accident timing) to estimate offsite consequences for the more significant, albeit still remote, core melt accidents, as described below.

Scenario Selection

The process of selecting sequences for analyses in the SOARCA project was the subject of considerable deliberation, discussion, and review. The central focus of this reassessment is to introduce the use of a detailed, best-estimate, self-consistent quantification of sequences 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 off-site consequence (MACCS2) code, including both improved site-specific and non-site specific code input and updated sequence-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 sequences.

SOARCA is intended to provide perspective on the likely (ie. best estimate) outcomes of a severe accident at a nuclear power plant. The updated SOARCA requantification of consequences might include consideration of those sequences important to risk as demonstrated by a full-scope level 3 Probabilistic Risk Assessment (PRA). In practice, that is not feasible since there are no current full scope level 3 PRAs generally available, considering both internal and external events, to draw upon. However, the preponderance of level 1 PRA information, combined with our insights on severe accident behavior, is available on dominant core damage sequences, especially internal event sequences. This information, combined with our understanding of containment loadings and failure mechanisms together with radionuclide release, transport and deposition, allow us to utilize core damage frequency (CDF) as a surrogate criterion for risk. Thus, for SOARCA we elected to analyze sequences with a CDF greater than 10^{-6} per reactor-year. In addition, we included sequences that have an inherent potential for higher consequences (and risk), with a lower CDF - 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 the adoption of these criteria, we are reasonably assured that the more probable and important core melt sequences will be captured.

All the sequences identified in the SOARCA study are significant in an absolute sense. The American Society of Mechanical Engineer's "Standard for Probabilistic Risk Assessment for Nuclear Power Plants," ASME RA-Sb-2005, which was endorsed by the staff in Regulatory Guide 1.200, defines a significant sequence, in part, as one that individually contributes more than 1 percent to the core damage frequency (CDF). A CDF of 10^{-4} per reactor-year is an acceptable surrogate for the quantitative health objectives contained in the Commission's Safety Goal Policy Statement [51 FR 28044] (see Appendix D of NUREG-1860). It then follows that the SOARCA sequence selection criterion of 10^{-6} is 1 percent of an acceptable CDF goal and the SOARCA sequences are consistent with previously issued regulatory guidance.

Another way to judge the impact of low-frequency events is to consider the increase in the latent cancer consequences that would be necessary to offset the lower frequency. Conceptually, an event with a larger radiological release could have greater latent cancer risk if the increase in the radiological release is larger than the decrease in frequency of the event. For example, assuming the accident timing remains the same and using an LNT risk assumption, a 10^{-8} per reactor year event must have a radiological release more than 10 times the magnitude of an event with a frequency of 10^{-7} per reactor year in order to pose greater latent cancer risk. Since we are including events with substantial volatile releases on the order of 10 percent, it is, practically speaking, not feasible to achieve greater latent cancer fatality risk by increasing the magnitude of the release by more than a factor of 10.

Other than the magnitude of the radiological release, a major impact on both early and latent cancer fatality risks is derived from the timing of the offsite release. In this respect, we have examined candidate SOARCA sequences with timing in mind, both the timing of core damage and the timing of containment failure. As part of this consideration, we addressed, for the Peach Bottom plant, an additional sequence, the short term station blackout (SBO), even though it did not satisfy our selection criterion. The short-term SBO frequency is roughly an order of magnitude lower than the long-term SBO ($3x10^{-7}$ per reactor-year versus $3x10^{-6}$ per reactor year); however, the short-term SBO has a more prompt radiological release and a slightly larger release over the same interval of time. Our initial qualitative assessment of the short-term SBO led us to conclude that it would not have greater risk significance than the long-term SBO. Because, while it was a more prompt release (8 hours versus 20 hours), the release was delayed beyond the time needed for successful evacuation. In order to demonstrate the points regarding risk versus frequency for lower frequency events, we nonetheless included a detailed analysis of the short-term SBO. Table 5 shows the results of that analysis, and it can be seen that the absolute risk is indeed smaller for the short-term SBO than for the long-term SBO. Table 6 shows the same trends for the Surry sequences, where the lower frequency sequences may have greater conditional risk but absolute risk is smaller than or equivalent to other higher frequency sequences.

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

We also have compared the SOARCA sequences against those identified as important to risk in NUREG-1150 for the Surry and Peach Bottom plants. Adjusting for the improvements in our understanding of phenomena due to the research completed since the NUREG-1150 study was completed (roughly 18 years ago), we have found that, with one exception, SOARCA addresses the more likely and important sequences identified in that landmark study. The one exception—a sequence identified in NUREG 1150 that has not been analyzed for the SOARCA project—involved an extreme earthquake that directly results in a large breach of the reactor coolant system (large loss-of-coolant accident [LOCA]), a large breach of the containment, and an immediate loss of safety systems. We conclude that this sequence is not appropriate for consideration as part of SOARCA for a number of reasons. Foremost, the state of quantification of such extreme and low-frequency seismic events is poor, considerable uncertainty exists in the quantification of the seismic loading condition itself, and a detailed soil-structure interactions analysis was not performed for the plant (and its equipment) response to the seismic loads. The analysis of the plant's components to the seismic acceleration—commonly referred to as fragility analysis—is a key component, and the lack of detailed analysis in this area makes current consideration of this event incompatible with the thrust of SOARCA, which is the performance of detailed, realistic analyses. Further, recent experience at nuclear plants in Japan strongly suggests that nuclear plant designs possess inherently greater capability to withstand the effects of extremely large earthquakes. In addition, it would not be sufficient to perform a nuclear plant risk evaluation of this event (even if it were currently feasible) without also performing an assessment of 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.

While we conclude that analysis of such an extreme earthquake that involves simultaneous failures of the reactor system, safety systems, and containment is not warranted as part of SOARCA, we believe that such events because of their potential for risk should be assessed as part of a separate future study. This future study, which will be integrated into the NRC seismic research program, will include the development of detailed mechanistic models for site-specific plant response as well as assessment of the nonnuclear seismic impacts on the general public.

In summary, SOARCA addresses the more likely (though still remote) and important sequences that are understood to compose much of the severe accident risk from nuclear plants. We 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.

Mitigation Measures

In preparation for the detailed, realistic modeling of accident progression and offsite consequences, the 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 for simulation of site-specific and scenario-specific emergency planning. Further, in addition to input for model development, licensees provided information from their own PRA on accident scenarios. Through table-top 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 the implementation of the available mitigation measures. This includes mitigation measures beyond those treated in current PRA models. Mitigation measures treated in SOARCA include Emergency Operating Procedures (EOPs), Severe Accident Management Guidelines (SAMGs), and 10 CFR 50.54(hh) mitigation measures. 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. Based on this verification and the previously discussed assessment of mitigation measures, including emergency operating procedures, severe accident management guidelines, and security-related mitigation measures, the staff concluded that all the identified scenarios could reasonably be mitigated.

Scenarios identified in SOARCA included both externally and internally initiated events. The externally initiated events frequently included events for which seismic, fire, and flooding initiators were grouped together. For the externally initiated events, the timeline of operator actions was developed assuming the initiator was a seismic event because the seismic initiator was judged to be the most severe initiator in terms of timing and with respect to how much equipment would be available to mitigate. Thus, there is some conservatism in attributing all of the event likelihood to a seismic initiator.

Accident Progression and Radionuclide Release

At the beginning of this project, an independent expert panel was assembled to review the proposed severe accident modeling approach of MELCOR to identify priority areas that would benefit from improvement prior to undertaking the SOARCA calculations. MELCOR is NRC's detailed mechanistic model that incorporates our best understanding of plant response and severe accident phenomenology. The SOARCA project team evaluated comments and recommendations made by the panel, and refinements or adjustments were made to the code and input files to improve the models.

MELCOR plant system models for Peach Bottom and Surry also were upgraded based on updated information from the licensees (e.g., system flow rates and actuation criteria). In addition, updated containment structural and leakage performance models were added to the MELCOR Peach Bottom and Surry models based on an extensive containment experimental research program conducted at Sandia National Laboratories that revealed concrete containments would experience an increase in leakage that would prevent catastrophic failure. With respect to Peach Bottom, improved modeling of drywell head leakage was incorporated. The use of MELCOR for SOARCA represents a significant and fundamental improvement over past consequence and risk studies.

The assessment of the mitigation measures was undertaken with support from integrated accident progression analyses using the MELCOR code. The MELCOR models were constructed based on the previously discussed conclusion regarding the effective implementation of mitigative measures. MELCOR analyses were used to both confirm the time available to take mitigation measures and to confirm that those measures, once taken, were adequate to prevent core damage or to 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.

To assess the benefits of the various mitigative measures (which were scenario specific) and to provide a basis for comparison to past analyses of unmitigated severe accident scenarios, the SOARCA project analyzed these same groups of scenarios assuming the event proceeded as unmitigated.

Offsite Radiological Consequences

An independent expert panel was assembled to review the proposed severe accident modeling approach of MACCS2 to identify areas that would benefit from improvement. MACCS2 includes atmospheric transport and dispersion (ATD) of the released radioactive material, emergency response of the population, and health effects from doses received._The SOARCA project team evaluated the comments and recommendations made by the panel team and made refinements or adjustments to the code and input files to improve the models. A major improvement made to the code was expansion to 64 radial directions for plume travel instead of 16. Several input value improvements were made as well, including: (1) use of short (1 hour long) plume segments, (2) risk coefficients from the 5th Committee on the Biological Effects of Ionizing Radiation (BEIR V), (3) a radionuclide inventory that reflects modern burnup practices, and (4) non-site specific parameters that reflect an expert elicitation by the NRC and the Commission of European Communities.

MACCS models for Peach Bottom and Surry are based on 1 year of hourly weather data from the licensees' meteorology towers and were updated to include site-specific population distributions for 2005. Also, site-specific public evacuation models were developed for each scenario based on the licensees' updated Emergency Preparedness programs and state emergency response plans to reflect the actual evacuation time estimates and road networks at Peach Bottom and Surry.

These public evacuation models also are more detailed in that they use multiple evacuating cohorts. A cohort is any population subgroup, such as schoolchildren, general public, and special needs individuals that moves or shelters differently from other population subgroups. Each cohort moves at a different time and speed and may have different sheltering characteristics that allow more realistic representation of shielding factors applied to the population. Cohorts modeled within the Emergency Planning Zone (EPZ) included the general public, school children, special facilities such as hospitals, and a nonevacuating cohort. The nonevacuating cohort of 0.5 percent of the public was used to represent individuals who do not follow the protective action recommendations. Research of large-scale evacuations has shown that only a small percentage of the public does refuse to evacuate (NUREG/CR-6864, 2005), and establishing this cohort helps to quantify this small population group.

A cohort outside the EPZ was used to represent a shadow evacuation. A shadow evacuation occurs when people evacuate from areas that are not under an evacuation order, and shadow evacuations are commonly observed in large-scale evacuations (NUREG/CR-6864, 2005). An estimate of 20 percent of the public in the area from 16 to 32 km (10 to 20 miles) from the plant was used to define the shadow evacuation. The shadow evacuation begins when an evacuation order is issued for residents of the EPZ. This 20 percent value was derived from a national telephone survey conducted to support NUREG/CR 6953, Volume II, "Review of NUREG-0654, Supplement 3, 'Criteria for Protective Action Recommendations for Severe Accidents'" (2008).

The offsite consequence analysis is based on the radioactive material release to the environment for the first 48 hours of the accident. The truncation of the release at 48 hours is intended to reflect the eventual termination of the release as a result of continually escalating mitigation action using both onsite and offsite resources. Because the release for the Surry long-term SBO does not start until 45 hours, consequence calculations for this sequence instead use a release truncation time of 72 to provide a basis for comparison to past analyses of unmitigated severe accident scenarios.

Offsite radiological consequence estimates are provided for each important scenario expressed as the average individual likelihood of an early fatality and latent cancer fatality conditional to the occurrence of a severe reactor accident and expressed as a risk metric factoring in the frequency of the scenario. The modeling of latent cancer fatality risk has been an issue of considerable controversy because evidence regarding risk is inconclusive in the low-dose region. To provide additional information on the potential range of health consequences, the SOARCA project has developed latent cancer risk estimates representing the range of health effects corresponding to the models proposed by international and national bodies. SOARCA assumed the LNT model and a range of truncation doses below which the cancer risk is not quantified. The LNT model is a basic assumption in many regulatory applications and is the model endorsed by the National Council on Radiation Protection and Measurements (NCRP) and the National Academies of Science (BEIR VII Report). Inclusion of dose truncation values is not meant to imply any NRC endorsement of a truncation value. Dose truncation values used for SOARCA included 10 mrem/year representing a small dose and the trivial dose below which the International Commission on Radiological Protection (ICRP) suggests avoiding summing doses, 620 mrem/year representing background radiation levels in the environment, and 5 rem/year with a 10 rem lifetime cap representing the Health Physics Society Position Statement in "Radiation Risk in Perspective," August 2004.

Results and Conclusions

Scenario Selection

The result of our scenario selection process, using updated and benchmarked Standardized Plant Analysis Risk (SPAR) models and the best available plant-specific external events information, was the identification of two major groups of accident scenarios. The first group, common to both Peach Bottom and Surry, was events, usually referred to as Station Black Out (SBO) scenarios, that include variations identified as short-term and long-term SBO. These scenarios involve a loss of all alternating current (ac) power, and the short-term SBO also involves the loss of turbine driven systems through loss of direct current control power or direct loss of the turbine system. The short-term SBO has a lower frequency because it involves more extensive system failures. These scenarios were typically initiated by some external events—fire, flood, or seismic initiators. The initiators were not always well differentiated in external events PRA. For the purpose of SOARCA analyses, it was assumed the SBO was initiated by a seismic event. Notwithstanding the SOARCA process, SBO scenarios are commonly identified as important contributors in PRA because of the common failure mode nature of the scenario and the fact that both containment safety systems and reactor safety systems are affected.

The second scenario group, which was identified for Surry only, was the containment bypass scenario. For Surry, two bypass scenarios were identified and analyzed—one involved an interfacing systems Loss of Coolant Accident (ISLOCA) due to an unisolated rupture of low pressure safety injection piping outside containment, and the other scenario involved a thermally induced steam generator tube rupture. The SPAR model frequency for the ISLOCA of $3x10^{-8}$/reactor-year falls below the SOARCA screening criteria for bypass events ($1x10^{-7}$/reactor-year). However, SOARCA analyses included this scenario because the licensee's PRA for Surry included an ISLOCA frequency of $7x10^{-7}$/reactor year and it has been commonly identified as an important contributor in PRA. The thermally induced steam generator tube rupture scenario occurs as a variant of an SBO scenario. This scenario also is generally understood to be an important potential contributor to risk in PRA. The scenarios are listed in Tables 1 and 2.

Table 1 Peach Bottom Scenarios Selected for Consequence Analysis

Scenario	Initiating Event	Core damage frequency (per reactor-year)	Description of scenario
Long-term SBO	Seismic, fire, flooding	$3x10^{-6}$	Immediate loss of ac power and eventual loss of control of turbine-driven systems due to battery exhaustion
Short-term SBO	Seismic, fire, flooding	$3x10^{-7}$	Immediate loss of ac power and turbine-driven systems

Table 2 Surry Scenarios Selected for Consequence Analysis

Scenario	Initiating Event	Core damage frequency (per reactor-year)	Description of scenario
Long-term SBO	Seismic, fire, flooding	$2x10^{-5}$	Immediate loss of ac power and eventual loss of control of turbine-driven systems due to battery exhaustion

Short-term SBO	Seismic, fire, flooding	2×10^{-6}	Immediate loss of ac power and turbine-driven systems
Thermally induced steam generator tube rupture	Seismic, fire, flooding	5×10^{-7}	Immediate loss of ac power and turbine-driven systems, consequential tube rupture
Interfacing systems LOCA[1]	Random failure of check valves	3×10^{-8}	Check valves in high-pressure piping fail open causing low-pressure piping outside containment to rupture, followed by operator error

Mitigation Measures

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 short-term SBO, the mitigation was sufficient to enable flooding of the containment through the containment spray system to cover core debris. The assessment of the mitigation measures was undertaken with support from integrated accident progression analyses using the MELCOR code. MELCOR analyses were used to both confirm the time available to take mitigation measures and to confirm 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. In addition, the release was truncated at 48 hours as a result of continually escalating mitigation actions, including containment and reactor building flooding.

Accident Progression and Radionuclide Release

An important result of the MELCOR accident progression analyses was the insight that accident progression in severe accidents proceeds much more slowly than earlier treatments indicated. The reasons for this are principally twofold—(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 and (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. In general, bounding approaches

[1] The licensee's PRA core damage frequency was 7×10^{-7}.

have been used in past simplified treatments using 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.

In the most likely accidents considered in SOARCA (assuming no mitigation)—the long-term SBO—core damage was delayed for 10 to 16 hours and reactor vessel failure was delayed for approximately 20 hours. Approximately 20 hours (BWR) or 45 hours (PWR) were available before the onset of offsite radiological release due to containment failure. In the most widely referenced siting study scenario (identified as the SST1 release), it was assumed that a major release occurs in 1½ hours. The SOARCA analyses showed that ample time is available for operators to take corrective action and for input from plant technical support centers even if initial efforts are assumed unsuccessful. Even in the case of the most rapid events (i.e., the unmitigated short-term SBO where core damage may begin in 1 to 3 hours), reactor vessel failure is delayed for roughly 8 hours allowing time for restoration of cooling and preventing vessel failure. In these cases, containment failure and radiological release is delayed for 8 hours (BWR) or 24 hours (PWR). For the bypass events, substantial delays occur or, in the case of the thermally induced steam generator tube rupture, the radiological release is shown by analyses to be substantially reduced. Tables 3 and 4 provide key accident progression timing results for SOARCA scenarios.

Table 3 Peach Bottom Accident Progression Timing Results

Scenario	Time to start of core damage (hours)	Time to lower head failure (hours)	Time to start of release to environment (hours)
Long-term SBO	10	20	20
Short-term SBO	1	8	8

Table 4 Surry Accident Progression Timing Results

Scenario	Time to start of core damage (hours)	Time to lower head failure (hours)	Time to start of release to environment (hours)
Long-term SBO	16	21	45
Short-term SBO	3	7	25
Thermally	3	7.5	3.5

induced steam generator tube rupture			
Interfacing systems LOCA	9	15	10

The SOARCA study also demonstrated that the magnitude of the radionuclide release is likely to be much smaller than used in past studies, again as a result of extensive research and improved modeling and as a result of integrated and more complete plant simulation. Some releases of important radionuclides such as iodine and cesium are predicted to be about 10 percent, but are more generally in the range of 0.5 to 2 percent. By contrast, the siting study used an iodine release of 45 percent and a cesium release of 67 percent. Figures 1 and 2 provide the radionuclide release results for iodine and cesium.

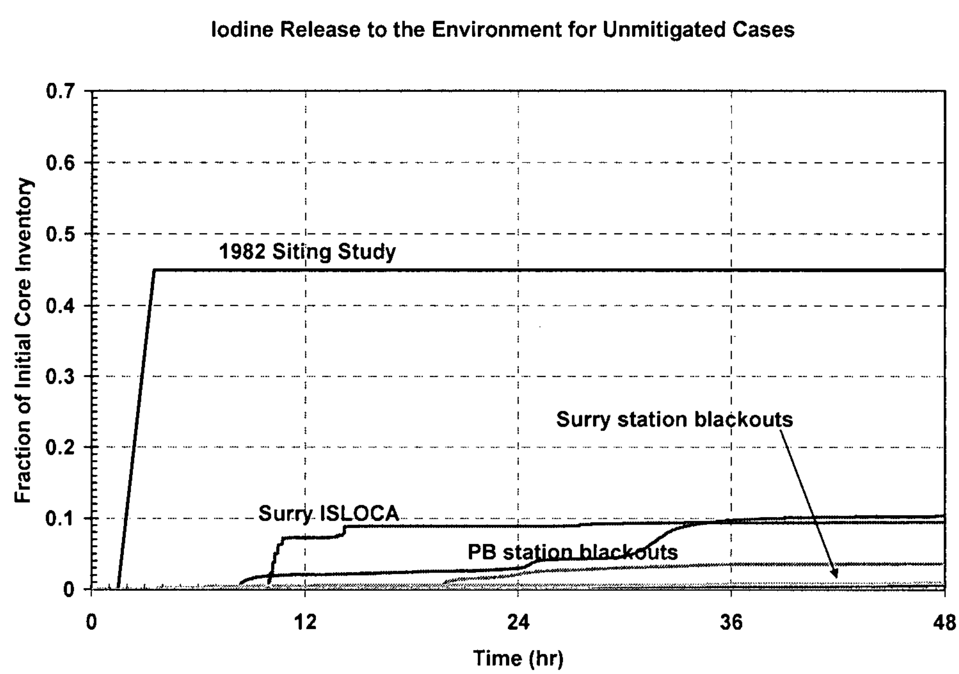

Figure 1. Iodine Releases to the Environment for SOARCA Unmitigated Scenarios

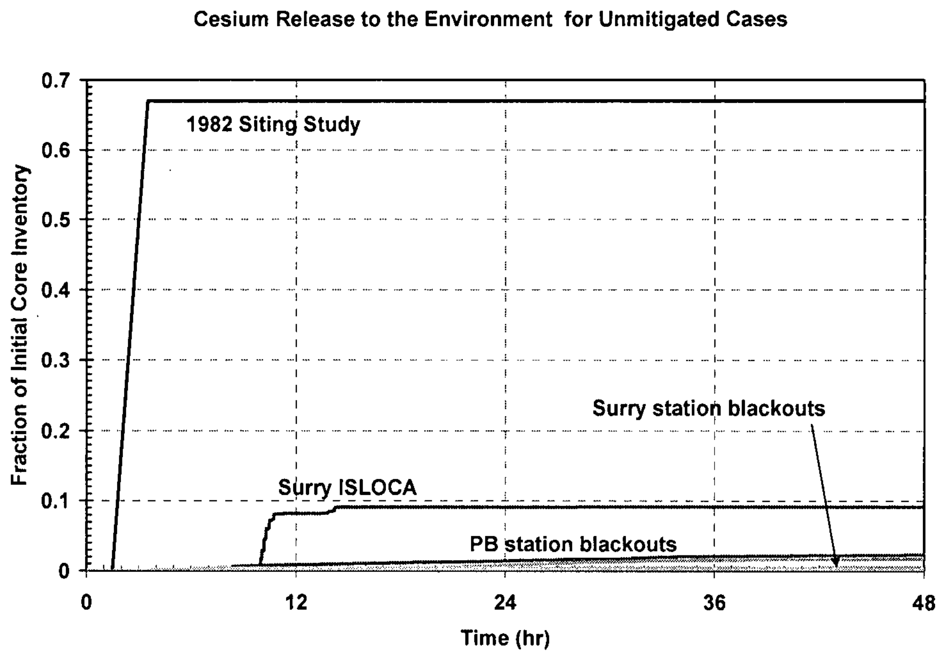

Figure 2. Cesium Releases to the Environment for SOARCA Unmitigated Scenarios

Sequences involving large early releases have influenced the results of past PRAs and consequence studies. For example, the siting study results were controlled by an internally initiated event with a large early release that was assigned a representative frequency of $1x10^{-5}$/year, based on knowledge available at the time. However, in the SOARCA study, no sequences with a frequency above $1x10^{-7}$/year resulted in a large early release, even considering external events and unsuccessful mitigation. This is a result of research conducted over the last 2 decades that has shown that phenomena earlier believed to lead to a large early release are of extremely low probability or physically unfeasible. This research was focused on phenomena that have been previously assumed to be prime contributors to severe accident risk, including direct containment heating and alpha mode failure.

The PWR SBO with a thermally induced steam generator tube rupture has in the past been believed to result in a large, relatively early release potentially leading to higher offsite consequences. However, MELCOR analysis performed for SOARCA showed that the release was small owing to thermally induced failures of other reactor coolant system components after the tube rupture. Also, the release was somewhat delayed; for the short-term SBO where no injection occurred at the start of the accident, the tube rupture and release began about 3.5 hours into the event. Further, core damage, tube rupture, and radiological release could be delayed for many hours if auxiliary feedwater were available even for a relatively short time period.

<u>Offsite Radiological Consequences</u>

The result of the accident progression and source-term analysis is that releases are delayed and of a diminished magnitude, Because of this and the realistic simulation of emergency response, essentially no early fatalities were predicted, as close-in populations were evacuated before or shortly after plume arrival.

Latent health effects calculated using any of the dose-response models referenced in this study are small in comparison to the Safety Goal. Much of the latent cancer risk for the close-in population was in fact derived from the relatively small doses received by populations returning to their homes in accordance with emergency planning guidelines. For example, for the Peach Bottom long-term SBO, about 70 percent of the latent cancer risk to individuals within 50 miles is from returning home. Here, the prediction of latent cancer risk, though very small, is strongly influenced by the relationship between low-dose health effects modeling and criteria for allowing return of populations.

Estimates of conditional (i.e., assuming the accident has occurred) individual latent cancer risk range from roughly 10^{-3} to 10^{-4}, using the LNT dose response model (other dose models result in lower or much lower conditional risk). If one also accounts for the probability of the severe accident itself, without successful mitigation (denoted as the absolute risk below), the risk to an individual for an important severe accident scenario is on the order of 10^{-9} to 10^{-10} per reactor year. These risk estimates are a million times smaller than the U.S. average risk of a cancer fatality of 2×10^{-3} per year. Tables 5 and 6 provide the risk estimates for individual SOARCA scenarios without successful mitigation using the LNT dose response model. The risk estimates are based on an assumed truncation of the release at 48 hours as a result of continually escalating mitigation actions, including containment and reactor building flooding.

Table 5 Peach Bottom Results for Scenarios Without Successful Mitigation and Assuming LNT Dose Response Model

Scenario	**Core damage frequency (per reactor-year)**	**Conditional risk of latent cancer fatality for an individual located within 10 miles**	**Absolute risk of latent cancer fatality for an individual located within 10 miles (per reactor-year)**
Long-term SBO	3×10^{-6}	2×10^{-4}	6×10^{-10}
Short-term SBO	3×10^{-7}	2×10^{-4}	7×10^{-11}

Table 6 Surry Results for Scenarios Without Successful Mitigation and Assuming LNT Dose Response Model

Scenario	**Core damage frequency (per reactor-year)**	**Conditional risk of latent cancer fatality for an individual located within 10 miles**	**Absolute risk of latent cancer fatality for an individual located within 10 miles (per reactor-year)**
Long-term SBO	2×10^{-5}	5×10^{-5}	7×10^{-10}
Short-term SBO	2×10^{-6}	9×10^{-5}	1×10^{-10}
Thermally induced steam generator tube rupture	5×10^{-7}	3×10^{-4}	1×10^{-10}
Interfacing systems LOCA	3×10^{-8}	7×10^{-4}	2×10^{-11}

To provide additional information on the potential range of health consequences, the SOARCA project has developed latent cancer risk estimates assuming the LNT (endorsed by NCRP) and a range of truncation doses below which the cancer risk is not quantified. Dose truncation values used for SOARCA included 10 mrem/year (representing ICRP), 620 mrem/year (representative background radiation) and 5 rem/year with a 10 rem lifetime cap (endorsed by HPS). Tables 7 and 8 show the results of sensitivity calculations for dose truncation values for background and the Health Physics Society position compared with LNT results. Using these truncation values makes the already small latent cancer fatality risk estimates even smaller, in some cases by orders of magnitude. Using the 10 mrem/year truncation value made a relatively small change in the latent cancer risk compared with the LNT model and, therefore, these results were not included in Tables 7 and 8.

SOARCA analysis included predictions of individual latent cancer fatality risk for 3 distance intervals, 0 to 10 miles, 0 to 50 miles, and 0 to 100 miles. The analysis indicated that individual latent cancer risk estimates generally decrease with increasing distance in large part due to plume dispersion and fission product deposition closer to the site.

As noted above, the SOARCA offsite consequence estimates are smaller than reported in earlier studies. For example, the 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 latent cancer fatality results, the exact basis for the Siting Study 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. However, given this uncertainty SOARCA does not make a direct comparison to the Siting Study latent cancer fatality estimate.

Table 7 Peach Bottom Results for Scenarios without Successful Mitigation for LNT and Alternative Dose Response Models

	Absolute risk of latent cancer fatality for an individual located within 10 miles (per reactor-year)		
Scenario	**Linear No Threshold**	**Background**	**Health Physics Society**
Long-term SBO	$6x10^{-10}$	$3x10^{-11}$	$5x10^{-12}$
Short-term SBO	$7x10^{-11}$	$6x10^{-12}$	$4x10^{-12}$

Table 8 Surry Results for Scenarios Without Successful Mitigation for LNT and Alternative Dose Response Models

Scenario	Absolute 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}	2×10^{-11}	2×10^{-14}
Short-term SBO	1×10^{-10}	1×10^{-11}	2×10^{-14}
Thermally induced steam generator tube rupture	1×10^{-10}	4×10^{-11}	3×10^{-12}
Interfacing systems LOCA	2×10^{-11}	8×10^{-12}	5×10^{-12}

ACKNOWLEDGEMENTS

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ACRONYMS

AC	Alternating Current
AFW	Auxiliary Feedwater
CCI	Core Concrete Interactions
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CST	Condensate Storage Tank
DC	Direct Current
EAL	Emergency Action Levels
ECCS	Emergency Core Cooling System
ECST	Emergency Condensate Storage Tank
EPZ	Emergency Planning Zone
ETE	Evacuation Time Estimate
FR	Federal Register
GE	General Emergency
HPCI	High Pressure Coolant Injection
HPI	High Pressure Injection
IPEEE	Individual Plant Examination – External Events
ISLOCA	Interfacing Systems Loss-of-Coolant Accident
LLNL	Los Alamos National Lab
LOCA	Loss Of Cooling Accident
LOOP	Loss Of Offsite Power
LPI	Low Pressure Injection
LTSBO	Long Term Station Blackout
MSIV	Main Steam Isolation Valve
NG	Noble Gas
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
OREMS	Oak Ridge Evacuation Modeling System
ORO	Offsite Response Organizations
PGA	Peak Ground Acceleration
PRA	Probabilistic Risk Assessment
PORV	Power operated relief valve
RCIC	Reactor Core Isolation Cooling
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RWST	Refueling Water Storage Tank
SAE	Site Area Emergency
SBO	Station Blackout

ADS
CRDHS
BEF
LBT
LNT
MWt Megawatts thermal
LPCS

ETE ICRP
CEG NCRP
EPA IAEA
DCF SAMA
BEIR SAMDA
LCF CST
UNSCEAR ERO
HPS EOP
WHO STCP

SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SOARCA	State-of-the-Art Reactor Consequence Analysis Project
SORV	Stuck Open Relief Valve
SPAR	Simplified Plant Analysis Risk
SRM	Staff Requirements Memorandum
SRV	Safety Relief Valve
STSBO	Short Term Station Blackout
TAF	Top of Active Fuel
TD-AFW	Turbine Driven Auxiliary Feedwater
UE	Unusual Event

1.0 INTRODUCTION

This document describes the NRC's state-of-the-art, realistic assessment of the accident progression, radiological releases and offsite consequences for important severe accident sequences. to The primary focus of Volume I is to provide the background and objectives of the study and summarizes the methods used to perform the analysis. The Executive Summary also summarizes the key results. Volumes ~~III~~ II and ~~IV~~ III discuss the detailed modeling practices and the plant-specific results.

1.1 Purpose of SOARCA

The evaluation of accident phenomena and offsite consequences of severe reactor accidents has been the subject of considerable research by the U.S. Nuclear Regulatory Commission (NRC), the nuclear power industry and the international nuclear energy research community. Most recently, with Commission guidance and as part of plant security assessments, updated analyses of severe accident progression and offsite consequences were completed utilizing the wealth of accumulated research~~, which are more detailed, integrated, and realistic than past analyses~~. These analyses are considerably more detailed, integrated, and realistic than past analyses. Further, as a result of past risk assessments and in response to the terrorist attacks of September 11, 2001, nuclear plants have made additional safety enhancements which reduce the risk of severe accidents as portrayed in earlier NRC assessments.

for selected dominant accident scenarios at full power

The objective of the State-of-the-Art Reactor Consequence Analyses (SOARCA) project is to develop ~~the~~ updated estimates of potential site-specific offsite consequences from severe accidents for operating nuclear power plants (NPPs). The Peach Bottom Atomic Power Station and the Surry Power Station were the first two plants selected to perform ~~risk-informed~~ consequence analyses. The licensee provided detailed information on the current plant designs and configurations including their existing and newly developed mitigative measures~~, to the extent practical~~. The analyses were performed using state-of-the-art thermal-hydraulic severe accident progression modeling, state-of-the-art consequence analysis modeling, and the latest emergency preparedness assumptions and criteria to assess the potential effects to public health and safety ~~in the unlikely event of a severe accident at an operating U.S. nuclear power plant~~. of selected severe accidents
The resultant health consequences were used to determine average probability of an individual dying from ~~a prompt~~ an early fatality and latent cancer fatality cancer that were conditional on the occurrence of a severe reactor accident. The results of these recent studies have confirmed and quantified what was suspected but not well-quantified — namely, that some past studies of ~~plant response and offsite consequences~~ radioactive release and early fatalities were conservative to the point that predictions were not useful for characterizing results, ~~communicating to the public~~ or guiding public policy. ~~The subsequent misuse and misinterpretation of these estimates further suggests that communication of risk attributable to severe reactor accidents should be based on realistic estimates of the more likely outcomes~~.

SOARCA

(and also that reflected the likelihood of the accident (absolute risk)

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 sites, based on the most current emergency preparedness (EP) and plant capabilities. It is also anticipated that the study will be a resource for improvements ~~and validation of~~ in modeling for future (PRA) studies.

Probabilistic Risk assessment

1.2 Background

In the first decade of nuclear power, the reactors were low power and of experimental designs. The fission product inventories and heat removal requirements of these low power reactors were much lower of those today. As newer designs approaching 500 MWt were developed, the Atomic Energy Commission (AEC) began serious studies of accidents and their consequences. Over the following 40 years, the AEC and later the NRC would produce a number of reports that examined the broad spectrum of reactor risk and consequence. Each study built upon the prior studies and added newer research and experience in design and operation of power plants to sharpen the models of nuclear accidents.

1.2.1 WASH-740, Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants, 1957

the study "hardly constituted more than an identification of the factors which have important, the best appraisal of these factors currently possible, and a rough approximation of the magnitude of the composite result."

An important technical input to establishing the indemnity provisions of the Price-Anderson Act was the report WASH–740 [1], which was prepared by Brookhaven National Laboratory and published by the AEC. Using what would later prove to be extremely pessimistic assumptions including a core meltdown with the release of fifty percent of the core fission products to the atmosphere, the worst case consequences of a 500 MWt reactor accident were estimated to be 3,400 early fatalities, 43,000 acute injuries, and a 7 billion (1957) dollar financial impact. There was a consensus among those involved in the WASH-740 study that the likelihood of a meltdown accident was low, but quantitative probability estimates could not be supported given the lack of operating plant experience. Similarly, the likelihood of containment failure (or bypass) given a meltdown accident was not quantified (or quantifiable, at the time). However, until 1966, the containment building was treated as an independent barrier, which should remain intact even if the core melted, thereby preventing any large release of radionuclides to the atmosphere. It was recognized that failure of the containment building and melting of the core could occur—for example, as a consequence of gross rupture of the reactor pressure vessel—but such events were not considered credible. Containment failure was not expected to occur simply because the core melted.

If this is to be included we need to explain this comes from a 1957 document. How could 50% of the core be released if containment failure was not a given even if the core melted.

1.2.2 WASH-1250, The Safety of Nuclear Power Reactors (Light Water-Cooled) and Related Facilities, 1973

Senator John Pastore requested a comprehensive assessment of reactor safety. The AEC's first response to this request was the WASH-1250 report [2], which was published in final form in July 1973. WASH-1250 provided factual information regarding the conservatisms applied in the design of nuclear power plants. It did not, however, address the likelihood or potential consequences of beyond-design-basis, that is, failures beyond those postulated under the single failure criteria.

I didn't see any place "above" that discusses this issue.

~~As indicated above, the~~ Radionuclide releases from fuel assumed in conservative design-basis LOCA analyses could only be realized if significant core melting occurred. Consequently, for a severe accident in which containment remained functional, the resulting offsite doses would be comparable to those conservatively calculated in the Safety Analysis Report for design-basis LOCAs. Yet the possibility remains of severe accidents in which containment is ~~either bypassed~~ or breached as a result of severe accident phenomena. Depending on the mechanism, location, and timing of containment failure, and the meteorological conditions, offsite doses could be substantially (100 times) worse than conservatively calculated for the design-basis LOCA. That is, the accidents with the greatest potential public consequences are uncontained severe accidents.

Loss of Coolant Accident

Is this a correcting or short comings of the document?

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

Because of shortcomings in WASH-1250
~~In this light~~, several questions had to be addressed in order to respond to Senator Pastore's request for a comprehensive assessment of reactor safety. What accidents could result in significant core damage and containment breach or bypass? How likely are these accidents? What would be their health and economic consequences? These are fundamental questions that WASH-1250 did not address. Such questions are now addressed in ~~probabilistic risk assessments~~ PRAs, but, at the time, ~~relevant~~ technical probabilistic estimates were quite limited in scope and highly subjective.

In the summer of 1972 the AEC initiated a major probabilistic study, the Reactor Safety Study. Professor Norman C. Rasmussen of the Massachusetts Institute of Technology served ~~(half-time)~~ as the study director. Saul Levine of the AEC served as ~~full-time~~ staff director of the AEC employees that performed the study with the aid of many contractors and consultants.

The team attempted to make a realistic estimate of the potential effects of light water reactor (LWR) accidents on the public health and safety. One BWR (Boiling Water Reactor), Peach Bottom Unit 2, and one PWR (Pressurized Water Reactor), Surry Unit 1, were analyzed in detail to estimate the likelihood and consequences of potential accidents. These plants were chosen because they were the largest plants of each type that were about to start operation.

The Study's
~~Its~~ stated purpose was to quantify the risks to the general public from commercial ~~NPP~~ nuclear power plant operation. This logically 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 [3] modeled scenarios leading to large radiological releases from each of two ~~types of commercial NPPs~~ plants. It considered highly complex scenarios involving success and failure of many and diverse systems ~~within a given scenario~~, as well as operator actions and phenomenological events. ~~These kinds of considerations were not typical of classical reliability evaluations. In fact~~ The to

and that went well beyond the normal analyses involved in then-existing design basis analyses.

~~address public risk, WASH-1400 needed to evaluate and classify many scenarios whose phenomenology placed them well outside the envelope of scenarios normally analyzed in any detail.~~

The team adapted methods previously used by the 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, ~~which~~ PRA is still utilized ~~, is called probabilistic risk assessment (PRA).~~ and has been adapted and improved upon in subsequent decades.

The team first identified events that could potentially lead to core damage. Event trees were then used to delineate possible sequences of successes or failures of systems provided to prevent core meltdown and/or the release of radionuclides. Fault trees were used to estimate the probabilities of system failures from available data on the reliability of system components. Using these techniques, thousands of possible core melt accident sequences were assessed for their occurrence probabilities. The public health and economic consequences of ~~the identified~~ representative groups of those severe accidents were estimated using computational models that were developed as part of the overall effort.

The insights gained from the Reactor Safety Study included: (a) melting of the reactor core does not necessarily result in an accident having large public consequences, (b) ~~the number of fatalities expected from the most likely course of events following a melting of a core is much smaller than those that commonly occur in accidents~~ Reactor risks are predicted to be much smaller than many other man-made and natural risks such as fires, explosions ~~and crashes of a commercial jet airplane~~, dam failures, air travel, chlorine, tornados, hurricanes, and earthquakes, and (c) ~~there are wide varieties of weather conditions and population densities where reactors are located and, when appropriate frequencies of occurrence are assigned,~~ these can cause potential accident consequences to increase by 100 to 1000 times; however, the probability ~~of such accidents could decrease by generally similar factors.~~ when appropriate frequencies of occurrence are assigned

It was assumed that there ~~are~~ were 100 power reactors and that ~~they all have~~ each has risks equal to the average risks for Surry ~~and Peach~~ Bottom. This assumption was not rigorously investigated. ~~In particular~~ the study stated as a limitation that it would not be appropriate to extrapolate the results beyond 100 reactors and 5 years. This limitation was based on the observation of continued attention to improved safety. The assumed improvement depended strongly on the continuing existence of competent and well supported regulatory and reactor safety research programs and reasonably conservative extrapolation of current practice. on average;

While the risks from nuclear power appeared to be very low, the Reactor Safety Study did indicate that core melt accidents were more likely than previously thought (approximately 5×10^{-5} per reactor year for Surry and Peach Bottom), and that light water reactor 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 and/or operator errors) that have the potential to cause core melt. In particular, the report indicated that, for the plants analyzed, accidents initiated by transients or small LOCAs were more likely to cause core melt than the traditional large design-basis LOCAs.

show the usefulness of mitigating events

In addition to providing some quantitative perspective on severe accident risks, WASH- 1400 provided other results whose significance has helped to ~~drive~~ the ~~increased~~ application of PRA in the commercial nuclear power arena. It showed, for example, that some of the more frequent, less severe initiating events ~~(IEs)~~ (e.g., "transients") lead to severe accidents at higher expected frequencies than do some of the less frequent, more severe ~~IEs~~ (e.g., very large pipe breaks). It led to the beginning of the understanding of the level of design detail that must be considered in PRA if the scenario set is to support useful findings (e.g., consideration of support systems and environmental conditions). Following the ~~severe core damage~~ core melt event at Three Mile Island (TMI) in 1979, application of these insights gained momentum within the nuclear safety community, leading eventually to a PRA-informed re-examination of the allocation of licensee and ~~regulatory (U.S. Nuclear Regulatory Commission)~~ NRC safety resources. ~~In the 1980s, this process led to some significant adjustments to safety priorities at NPPs; in the 1990s and beyond, regulation itself is being changed to refocus attention on areas of plant safety where that attention is more worthwhile.~~

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

Following the TMI accident, NRC contracted with Sandia National Laboratory to develop a technical guidance report for siting future reactors [4]. Guidance was requested regarding (1) criteria for population density and distribution surrounding future sites, and (2) standoff distances of plants from offsite hazards. (SST)

considered 92 sites, covering both existing and proposed those those only

The ~~92 plant scope of~~ this scope study was so large that rather than model the release from ~~each plant~~ those separately, 5 types of accidents would be imposed on each plant. The accidents or "siting source term events" would be derived from the previous Reactor Safety Study ~~(WASH-1400)~~ and each SST event would be assumed identical regardless of the reactor size or plant design. Although the absolute numerical results may be questionable due to the arbitrary source terms, the relative impact of population density, weather, and evacuation times would be apparent for every site in the United States. that might exist at a particular site. different types of plants

SST1 – Severe core damage. Loss of all safety systems and loss of containment after 1.5 hours.

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

SST3 – Severe Core damage. Containment systems function but small containment leakage (1 % per day) after 1 ~~hr~~ hour.

SST4 – Modest core damage. Containment systems function but small containment leakage after ½ ~~hr~~ hour.

SST5 – Limited core damage. Containment functions as designed with minimal leakage.

The results for most of the 92 reactor ~~plants~~ sites were similar due to a low population density. Using the SST1 model with a population density of 50 persons per square mile resulted in 47 to 140 locations may be included.

and assumed that all doses, no matter how small, were able to cause cancer (the linear no threshold (LNT) model).

early fatalities and 730 to 860 latent cancers. For the more realistic release represented by SST2 events, the mean values from typical plants were zero early fatalities and 95-140 latent cancers.

For high population density sites the ~~consequences~~ latent cancer fatalities were higher, although not proportionally higher and this is a result of several factors. First, the study computed latent cancers based on the aggregate population dose ~~rather than the individual dose~~. As the distance from the accident site increases ~~the area increases with the square of the distance and in turn may include~~ higher population density. Thus, a di~~sproportionate~~ fraction of the latent cancers were derived from large distant populations that had received small individual doses. Secondly, it was assumed in CRAC2 (and in the present study) that long-term off-site response actions would be taken to reduce the doses. For instance, the amount of land removed from public use (interdicted) was found to be sensitive to the release fraction of cesium, while the total population dose was less affected. The factor that affected the split between the interdicted land and the population dose and meteorology was the criterion that was used to define interdiction (the habitability criterion). ~~The~~ highest consequence site using the SST1 ~~model~~ with a NRC population density resulted in a latent cancer increase of 0.06% over normal incidence. For the more realistic release represented by SST2 events, the same location resulted in a latent cancer increase of 0.004% increase over normal incidence.

source term — is significant — New York city — The effect of small doses (<0.1Sv, <10Rem) on human population remains uncertain to this day, in spite of extensive research on the subject.

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

NUREG-1150 [5] 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 significantly enhanced the understanding of risk at nuclear power plants, and is considered a significantly updated and improved revision to the Reactor Safety Study. A major improvement was the specific inclusion of an uncertainty estimate for the core damage frequency and source term portions of the study, but not for the off-site consequence portion. The uncertainty estimate was based on extensive use of expert elicitation.

The five nuclear power plants analyzed in NUREG-1150 are:

- Unit 1 of the Surry Power Station, a Westinghouse-designed three-loop reactor in a sub atmospheric containment building, located near Williamsburg, Virginia;
- Unit 1 of the Zion Nuclear Power Plant, a Westinghouse-designed four-loop reactor in a large, dry containment building, located near Chicago, Illinois;
- Unit 1 of the Sequoyah Nuclear Power Plant, a Westinghouse-designed four-loop reactor in an ice condenser containment building, located near Chattanooga, Tennessee;
- 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, Pennsylvania; and

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

for the two plants used in the present study

The various accident sequences that contribute to the core damage frequency from internal initiators can be grouped by common factors into categories. NUREG-1150 uses the accident categories depicted in **Table 9** below: station blackout, anticipated transients without scram, other transients, reactor coolant pump seal LOCAs, interfacing system LOCAs, and other LOCAs. The selection of such categories is not unique, but merely a convenient way to group the results.

Table 9 Summary of Core Damage Frequency from NUREG-1150

Plant Name	Internal Initiators							External Initiators
	SBO	~~Fire & Seismic~~	ATWS	TRANS	SG/IF Sys	LOCA	Core Damage Total/yr	Fire & Seismic
Surry	2.2E-5	~~2.6E-5~~	1.6E-6	2.0E-6	3.4E-6	6.0E-6	4.0E-5	2.6E-5
Peach Bottom	2.2E-6	~~2.3E-5~~	1.9E-6	1.4E-7	-	2.6E-7	4.5E-6	2.3E-5

2.2
.16
.20
.34
.60
3.50
.5
4.0

1.3 Objective of SOARCA

The ~~basic approach for~~ objective of the SOARCA project is to utilize ~~the~~ a self-consistent, integrated modeling of accident progression and offsite consequences, drawn from current best practices modeling, to estimate offsite consequences for important classes of events. This will be accomplished by modeling accident progression (reactor and containment thermal-hydraulic and fission product response), which is embodied in the MELCOR code, coupled with modeling offsite consequences, in the MACCS2 code, in a consistent manner (e.g., accident timing) and with improved input in important areas. Selection of the events for analysis was based on a consideration of insights from past and current PRAs and from research on accident behavior and failure modes important to latent cancer fatality and early fatality risks. Selection of events for quantification also ~~properly~~ included probability in order to focus on more likely and important contributors. It is believed that more can be learned ~~at this juncture~~ by focusing on a relatively few important events and quantifying the plant and offsite response rigorously and realistically than by approximate modeling of many events, including extremely rare events. This approach of focusing on a relatively few, but important events, also allows us to ~~efficiently and~~ explicitly address the benefits of additional mitigation in further reducing the likelihood of core damage and offsite consequences. ~~The off-site consequence analyses were performed on a site-specific basis (reflecting site-specific population distributions, weather, and EP) and also included improved understanding of non-site specific input.~~

Selection of events considered individual plant examinations (IPEs), individual plant examinations of external events (IPEEEs), standardized plant analysis risk (SPAR) models, and NUREG-1150 risk studies. Information related to system and procedural plant improvements that have been incorporated as part of the industry's response to the NRC's security initiatives, as well as necessary plant information was included in the scenario selection evaluation and incorporated in plant modeling.

1.4 Scope

1.4.1 SOARCA and Full Scope Level 3 PRA

The process of selecting sequences for analyses in the SOARCA project was the subject of considerable deliberation, discussion, and review. The central focus of this reassessment is to introduce the use of a detailed, best-estimate, self-consistent quantification of sequences 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 off-site consequence (MACCS2) code, including both improved site-specific and non-site specific code input and updated sequence-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 sequences.

as well as full power and low power and shutdown events

SOARCA is intended to provide perspective on the likely (ie. best estimate) outcomes of a severe accident at a nuclear power plant. The updated SOARCA requantification of consequences might include consideration of those sequences important to risk as demonstrated by a full-scope level 3 ~~Probabilistic Risk Assessment~~ (PRA). In practice, that is not feasible since there are no current full scope level 3 PRAs generally available, considering both internal and external events, to draw upon. However, the preponderance of level 1 PRA information, combined with our insights on severe accident behavior, is available on dominant core damage sequences, especially internal event sequences. This information, ~~combined with~~ our understanding of containment loadings and failure mechanisms, ~~together with~~ radionuclide release, transport and deposition, allow us to utilize core damage frequency (CDF) as a surrogate criterion for risk. Thus, for SOARCA we elected to analyze sequences with a CDF greater than 10^{-6} per reactor-year. In addition, we included sequences that have an inherent potential for higher consequences (and risk), with a lower CDF - 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 the adoption of these criteria, we are reasonably assured that the more probable and important core melt sequences will be captured.

level 1

our understanding

1.4.2 Multiple Units at One Site

Certain initiating events have the potential to affect multiple units at a given site or multiple sites. For example, on August 14, 2003, a widespread loss of the U.S. electrical power grid (blackout) resulted in loss of offsite power events at nine U.S. commercial nuclear power plants located at six different sites. An earthquake could similarly affect multiple units at the same site. Most PRAs developed to date do not explicitly consider multi-unit accidents because current NRC policy is to apply the Commission's Safety Goals (51 FR 28044) [6] and subsidiary risk

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However the possible need to assess multi-unit accidents was referred to the NRC's generic issues program

acceptance guidelines (see Regulatory Guide 1.174) on a "per reactor" basis [7]. Therefore, no multi-unit accident scenarios were selected for the SOARCA project.

1.4.3 Mitigated and Unmitigated Cases

If the mitigative measures were

An important objective of the SOARCA project was to assess the impact of severe accident mitigative features and reactor operator actions to mitigating the accident. This was done by evaluating in detail the operator actions and equipment which may be available ~~(including 10 CFR 50.54(hh) equipment)~~ to mitigate the specific accident sequences to determine if time was available to take corrective action and whether the equipment itself would be available given the sequence. These mitigative measures analyses were qualitative, sequence-specific systems and operational analyses based on licensee identified mitigative measures from Emergency Operating Procedures (EOPs), Severe Accident Mitigation Guidelines (SAMGs), and other severe accident guidelines, ~~that are~~ applicable to, and determined to be available during a scenario. ~~whose~~ availability, capability and timing ~~was~~ utilized as an input into the MELCOR analyses. A limitation to ~~this approach is~~ that a comprehensive human reliability assessment has not been performed to quantify the probabilities of plant personnel succeeding in implementing these measures and the likelihood of success or failure is unknown. However, the NRC has issued 10 CFR 50.54(hh) requiring plant licensees to possess the equipment, develop the strategies and have trained personnel to implement these mitigative measures. For sequences in which it was determined that mitigative measures would be taken, detailed accident progression analyses were conducted to assess the efficacy of those measures, given proper implementation. For such sequences accident progression and offsite consequence analyses were also performed assuming the mitigative measures were not taken, in order to demonstrate the relative importance/significance of those measures.

the measures were

(mitigated cases)

(unmitigated cases)

required post-9/11 and to

therefore

For those scenarios within the scope of SOARCA, applicable mitigative measures that are potentially available (not eliminated by initial conditions) were identified. The systems and operations analyses were based on the initial conditions and anticipated subsequent failures to:

(that is)

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

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

1.4.4 Key Assumptions

used in the development of the accident progression and consequence analysis for the SOARCA project.

~~In the development of the accident and consequence analysis for the SOARCA project~~, The concepts, applications, and parameters are identified in detail in the applicable report sections. ~~Assumptions are identified throughout the report in the appropriate sections that address the analysis.~~ Some of the overarching assumptions used in the SOARCA project are identified below.

However,

1.4.4.1 Accident Progression Analysis

The progression of events in a severe accident has uncertainties in phenomenological responses, equipment performance, and operator actions. An independent expert panel was assembled to review the proposed base case approach as well as to identify other areas of MELCOR modeling ~~to identify areas~~ that would benefit from improvement. The discussion of the best estimate modeling practices for MELCOR is summarized in ~~Volume II~~ Reference XX.

1.4.4.2 Consequence Analysis

Site-specific ~~Values~~ values used ~~in the documentation provided for the sites were used whenever possible~~ are discussed in the volumes for the specific sites (Vol II for Peach Bottom, Vol III for Surry). This includes evacuation time estimates, the population surrounding the plant, and other parameters as appropriate.

- Emergency response was modeled for each scenario for each plant. Cohorts (groups of population that have similar response) representing a non-evacuating population, school children, special needs population, a shadow evacuation (evacuation from areas not under an evacuation order) and the general population in the EPZ. These are discussed in detail in Volumes ~~III~~ II and ~~IV~~ III.

- Ingestion dose via the food pathway was bypassed, reflecting a modeling assumption that food supplies within the U.S. are sufficient so that eating contaminated food following an accident is unnecessary.

- Other, non-site specific parameters were obtained from a series of studies conducted by the NRC and the Commission of European Communities to develop credible and traceable uncertainty distributions for important input parameters. The specific values used in the best estimate case were the medians of the distributions that were developed by the NRC from the results of the studies.

1.5 Uncertainty Analysis

As part of SOARCA a number of sensitivity studies have been performed to examine the issues associated with accident progression, mitigation and offsite consequences. These sensitivity studies were performed to examine specific issues and to assure the robustness of the conclusions. Single sensitivity studies, however, can not answer questions concerning the overall uncertainty of the consequence or risk estimates attributable to the accident progression and offsite consequence modeling. ~~, which is the primary focus of the SOARCA project.~~

Therefore, a future ~~work~~ effort in the SOARCA project ~~is~~ will be to quantify the integrated uncertainty inherent in the analyses of accident progression and offsite consequences, focusing on the uncertainty in timing and magnitude of the radiological release and the offsite health effects. The uncertainty study will consider both aleatory (random) and systemic contributors to uncertainty. Primarily, the uncertainty study wi~~ll model parameter uncertainty and~~ will use parameter distributions to capture uncertainty ~~in~~ of phenomenological behavior as well as system, sequence or offsite response aspects ~~to~~ of uncertainty. The uncertainty analysis will be performed for one of the accident scenarios, which will be selected considering both the conditional risk and absolute risk estimated from the best estimate analysis.

1.6 Structure of NUREG XXXX and Supporting Documents

The structure of the NUREG is in multiple volumes. Volume I is the introduction to the SOARCA project and the methods and approaches used in the study. ~~Volume II contains the best practices for MELCOR application and is intended to be a ready resource for the severe accident modeling community.~~ Volumes ~~III~~ and ~~IV~~ contain the plant specific SOARCA results for the Peach Bottom and Surry plants, respectively. Additional plant-specific volumes will be added as volunteer-plants are identified and assessed. In addition, compact discs (CDs) are provided for each site. Each CD contains the input files for both MELCOR and MACCS2, so that, in the future, every input value will be traceable.

II V

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3.0 METHODS USED FOR MITIGATIVE MEASURES ASSESSMENT

A fundamental objective of SOARCA is to develop state-of-the-art analyses of accident progression, radiological release, and offsite radiological health consequences for risk-important severe accident sequence groups. 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 how available PRA information sources including the NRC's SPAR models, licensees' PRA models, NUREG-1150 and additional expert judgment were used to identify risk-important sequence groups leading to core damage and containment failure or bypass. This section describes the methods used to determine what mitigation measures would be available and the associated timing to implement. This includes mitigation measures beyond those treated in current PRA models. Mitigation measures treated in SOARCA include the licensee's emergency operating procedures (EOPs), severe accident management guidelines (SAMGs), and 1~~0 CFR 50.54(~~hh) mitigation measures. 1~~0 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 mitigation capability. It is expected that these measures would be implemented by the licensee's emergency response organization in accordance with the approved emergency plan. ation

post-9/11 the post-9/11

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 for simulation of site-specific and scenario-specific emergency planning. Further, in addition to input for model development, licensees provided information from their own PRA on accident scenarios. Through table-top 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. S

The licensee input for each scenario was used to develop timelines of operator actions and equipment lineup or setup times for the implementation of the available mitigation measures. ~~This includes mitigation measures beyond those treated in current PRA mode~~ls. ~~Mitigation measures treated in SOARCA include Emergency Operating Procedures (EOPs), Severe Accident Management Guidelines (SAMGs), and 10 CFR 50.54(hh) mitigation measures. 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.

Move to next page - already said above!

Scenarios identified in SOARCA included both externally and internally initiated events. The externally initiated events frequently included events for which seismic, fire, and flooding initiators were grouped together. For the externally initiated events, the timeline of operator actions was developed assuming the initiator was a seismic event because the seismic initiator was judged to be the most severe initiator in terms of timing and with respect to how much equipment would be available to mitigate. Thus, there is some conservatism in attributing all of the event likelihood to a seismic initiator.

3.1.1 Sequence Groups Initiated by External Events

The PRA screening identified the following sequence groups that were initiated by external events and met the SOARCA screening criteria of $1x10^{-6}$/reactor-year for containment failure events and $1x10^{-7}$/reactor-year for containment bypass events:

- Peach Bottom long-term station blackout – $1x10^{-6}$ to $5x10^{-6}$/reactor-year average $3x10^{-6}$
- Surry long-term station blackout – $1x10^{-5}$ to $2x10^{-5}$/reactor-year average $2x10^{-5}$
- Surry short-term station blackout – $1x10^{-6}$ to $2x10^{-6}$/reactor-year average $2x10^{-6}$
- Surry short-term station blackout with thermally induced steam generator tube rupture – $1x10^{-7}$ to $8x10^{-7}$/reactor year $5x10^{-8}$ not introduced until this time

These ~~sequence groups were initiated by a seismic, fire, or flooding event.~~ The mitigation ~~measures assessment for each of these sequence groups was performed assuming the~~ initiator was ~~a seismic event, because it was judged to be limiting in terms of how much~~ equipment ~~would be available to mitigate~~. Fewer mitigation measures are expected to be available for a seismic event than for an internal fire or flooding event. For these sequence groups, the seismic PRAs provided information on the initial availability of installed systems. Based on the estimated level of plant damage, the availability of ~~10 CFR 50.54(hh)~~ mitigation measures, their implementation time, and the timing and effectiveness of the emergency response organization support (e.g., in the Technical Support Center and Emergency Operating Facility) was evaluated. Post 9/11

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 systems RCIC and TDAFW are important mitigation measures. BWR SAMGs include starting RCIC without electricity to cope with station blackout conditions, ~~This is~~ known as RCIC black start. ~~10 CFR 50.54(hh)~~ mitigation measures have taken this a step further and also include long-term operation of RCIC without electricity (RCIC black run), using a portable generator to supply indications such as reactor pressure vessel 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 long-term station blackout sequence groups, RCIC and TDAFW can be used to cool the core until battery exhaustion. After battery exhaustion, black run of RCIC and TDAFW can be used to continue to cool the core. MELCOR calculations are used to demonstrate core cooling under these conditions. the post-9/11

Seismic PRAs for Peach Bottom and Surry do not describe general plant damage and accessibility. ~~The external events PRA does not describe general plant damage and accessibility following a seismic event.~~ The damage was assumed to be widespread and accessibility to be difficult, consistent with the unavailability of many plant systems. The seismic event was assumed to fail the Condensate Storage Tank in the Peach Bottom long-term station blackout, which is the primary water reservoir for RCIC. Consequently, RCIC must be initially supplied from the torus. MELCOR calculations showed that several hours would be available before torus temperature and pressure conditions precluded this. However, this would provide sufficient time to identify or arrange for another water reservoir for RCIC, such as the Peach Bottom cooling tower basin (a large low lying reinforced concrete structure). For the Surry long-term station blackout, the TDAFW system and the Emergency Condensate Storage Tank were not expected to fail. Consequently, the cooling water was supplied to the steam generators for ~~RCS~~ heat removal. It was assumed that eventually operators would provide make-up to the E~~mergency Condensate~~ S~~torage~~ T~~ank~~. For the Surry short-term station blackout, the ECST was assumed to fail and an alternative reservoir was assumed to be available by 8 hours; this could be achieved by using a fire truck or portable pump to draw from the river.

that is important to survive the seismic event (ECST)

from the RCS

not which ed

Also, for the Surry short-term station blackout, the low pressure injection and containment spray safety-related piping were judged not likely to fail. This judgment was primarily based on NUREG/CR-4334, ~~entitled~~ "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants" [11], ~~to~~ help extrapolate the potential viability of safety-related piping after a 1.0 g event. This conclusion also considered other related studies including a German study 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 system. Licensee staff estimated that transporting the pump and connecting it to plant piping takes about two hours. However, for the short-term station blackout this mitigation measure was estimated to take 8 hours due to the higher level of damage. Since 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 uncertainty in availability and accessibility for mitigation measures.

the Post-9/11 therefore,

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. At the time of the Surry site visit, the licensee had their portable equipment and the site fire truck onsite in a structure away from the containment. H~~ence~~, it was believed that portable equipment could be accessed and deployed for the seismic conditions evaluated in SOARCA. At the time of the Peach Bottom site visit, the licensee had not procured the required portable equipment.

of the time necessary

~~Time~~ Estimates to implement individual mitigation measures were provided by licensee staff for each sequence group based on scenario descriptions provided by the NRC. The time estimates take into account the plant conditions following the seismic event. Also, for portable equipment at Surry, the time estimates reflect exercises run by licensee staff that provided actual times to move the equipment into place. The time estimates for ~~manning~~ staffing the Technical Support Centers (TSC) and the Emergency Operating Facilities also were provided by licensee staff and reflect the possible effect of the seismic event on roads and bridges.

The mitigation measures assessment noted the possibility of bringing in equipment from offsite (e.g., fire trucks, or pumps and power supplies from sister plants or from contractors, ~~external spray systems~~), but it did not quantify the types, amounts, and timing of this equipment arriving and being implemented. Additional information on equipment available offsite and time estimates for transporting this equipment is available in Section 3.2.

Evaluating the effectiveness of external water spray using conventional firefighting equipment to scrub an ongoing fission product release was not evaluated in SOARCA. This evaluation is being performed in a separate study. did we consider this? If not, why not?

No multi-unit accident sequences were selected for the SOARCA project. Therefore, the mitigation measures assessment for external events was performed assuming that the operators only had to mitigate an accident at one reactor, even though Peach Bottom and Surry are two-unit sites. Also, at the time that the MELCOR models were developed for SOARCA, Surry Unit 1 had an opening in the reactor cavity wall and Surry Unit 2 did not. The MELCOR model for the Surry reactor includes an opening in the reactor cavity wall. Now Surry 2 has an opening? What does this have to do with mitigation measures?

3.1.2 Sequence Groups Initiated by Internal Events

The PRA screening identified the following sequence groups that were initiated by internal events and met the SOARCA screening criteria of $1x10^{-6}$/reactor-year for containment failure events and $1x10^{-7}$/reactor-year for containment bypass events:

- Surry interfacing systems LOCA – $7x10^{-7}$/reactor-year (licensee PRA), $3x10^{-8}$/reactor-year (SPAR)
- Surry spontaneous steam generator tube rupture – $5x10^{-7}$/reactor-year

These sequence groups result in core damage ~~as a result~~ because of assumed operator errors. For the interfacing systems LOCA, the operators fail to refill the RWST or cross-connect to the unaffected unit's RWST. For the spontaneous SGTR, the operators fail to 1) isolate the faulted SG, 2) depressurize and cooldown 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 Volume IV subsequently we don't know if the models did or did not? Surely one can tell –

left a voice mail for Marks 10/30/09

estimated 3 hours until the RWST is empty and 10 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 estimated could be longer for smaller break sizes. 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 the operators would correctly respond to the events. These time estimates included consideration of 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 ~~manned~~ staffed and operational by 1 to 1.5 hours into the event.

I thought that there was an orifice in the line so that LOCAs small not an issue

The mitigation measures assessment for internal events also included ~~10 CFR 50.54(hh)~~ post 9/11 mitigation measures, but these measures 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 sequence group as exceeding the SOARCA screening criterion of 1×10^{-6}/reactor-year. However, an inappropriate modeling assumption was subsequently found in the SPAR model, and the sequence group frequency was determined to be below the SOARCA screening criterion. However, by the time the issue was discovered, the mitigation measures assessment and the MELCOR analysis were complete. The MELCOR analysis described in Volume III demonstrated that this sequence group did not result in core damage, even without crediting 10 CFR 50.54(hh) mitigation measures, contrary to the more conservative treatment in SPAR. The mitigation measures assessment and the MELCOR analysis are described in this report to further demonstrate of the benefit of best-estimate integral accident progression analysis.~~

why bring this up? was a result of a mistake

3.2 Unmitigated Scenarios - Truncation of Releases

The assessment of mitigation measures, including emergency operating procedures, severe accident management guidelines, and security-related mitigation measures, led the project staff to conclude that all of the identified severe accident scenarios could be mitigated. 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 staff also analyzed the scenarios ~~conservatively~~ assuming the events proceed as unmitigated by available onsite mitigation measures and lead ultimately to an offsite release. ~~This NUREG refers to these~~ these scenarios are referred to here as unmitigated scenarios, because they are not effectively mitigated in the short-term by onsite resources. For these unmitigated scenarios, the project staff performed analysis to estimate the time by which offsite resources would be brought onsite and implemented to truncate the long-term revaporization release of fission products from the containment and other plant buildings.

The expected response to a severe nuclear power plant accident was reviewed to provide a basis for truncating the accident release. There are a multitude of resources available at the state, regional and national level that would be available to mitigate a NPP accident. The staff reviewed available resources and emergency plans and determined that adequate mitigation measures could be obtained within 24 hours and fully implemented within 48 hours for the onset of the mitigating event.

The National Response Framework (NRF) would be implemented in response to a severe nuclear power plant accident to coordinate the national level response. Under the NRF, DHS would be the coordinating agency and NRC would be a cooperating agency. The NRF is exercised periodically and improved based upon lessons learned. The NRC has an extensive, well-trained and exercised emergency response capability that would support, and under unusual circumstances, direct licensee efforts. The NRC has onsite inspectors that are available to provide first hand knowledge of accident conditions. 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 NRC would also activate the incident response team at headquarters. The focus of the NRC response is to ensure that public health and safety is protected and to assist the licensee with the response by coordinat~~ing~~ national ~~assest~~. the response

post-9/11

Both Surry and Peach Bottom are supported by a remote EOF. The ERO at the EOF has access to fleet-wide 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 a full onsite and offsite exercises biannually where response to severe accidents and coordination with offsite response organizations is demonstrated for and inspected by 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.

Significant resources would be made available to the site to mitigate the accident. While these efforts would be ad hoc, knowledgeable personnel and an extensive array of equipment would be available and ~~as such~~ were considered in the conclusion that radiological releases would be truncated within 48 hours.

working with (DHS) to

The Department of Homeland Security

?

The RIS know more (have more firsthand) information than the plant staff?

Emergency Response Official?

MELCOR

4.0 SOURCE TERM ANALYSIS

The source term is defined as the quantity, timing, and characteristics of the release of radioactive material to the environment following a postulated severe accident. The NRC has defined, calculated, and used source terms for a variety of research and regulatory activities. Two uses include (a) siting and regulatory applications and (b) probabilistic risk or consequence assessments. Two source terms used for regulatory applications include TID-14844 [12] and the alternate source term [13]. In contrast to the definition above, the regulatory source terms are releases to the containment, which are available for release to environment. The second use of the source term is an assessment of health consequence risks from severe accidents. There are many significant examples of the latter application i~~ncluding the Sandia Siting study~~ [4], N~~UREG-1150 [5], and SOARCA.~~ ~~In the Sandia Siting Study,~~ the postulated frequency distribution of five prescri~~bed source~~ terms of increasing severity were defi~~ned and~~ used to calculate the health ~~consequences~~ and risk. In NUREG-~~1150~~, a comprehensive, plant-specific evaluation ~~of all~~ significant ~~source terms were calculated using event tree models and~~ then ~~grouped into a much larger but manageable number of characteristic source terms to~~ calculate the health ~~consequences and risk~~. In the present SOARCA study, individual scenario source terms are evaluated using MELCOR code calculations and then evaluated for health consequences.

As discussed in Section 1.2

severe accident

Some background in key studies for regulatory and probabilistic applications is described in Section 4.1 below. Figure 4 shows a timeline of key events and NRC studies in the evolution of ~~nuclear/safety~~ technology. The key source terms studies cited in the timeline that preceded the SOARCA program are shown in the figure and discussed in Section 4.1 below. Next, a history of the severe accident source term codes developed by the NRC is described in Section 4.2. The MELCOR code is the culmination of the NRC research and code development of severe accident ~~phenomena for~~ source terms ~~evaluations~~. The scope of the MELCOR code and the relevant experimental programs supporting its advanced modeling capabilities are summarized in Section 4.3. The MELCOR modeling approach used in the SOARCA analyses is presented in Section 4.4. The MELCOR modeling approach 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.

into the phenomenology of

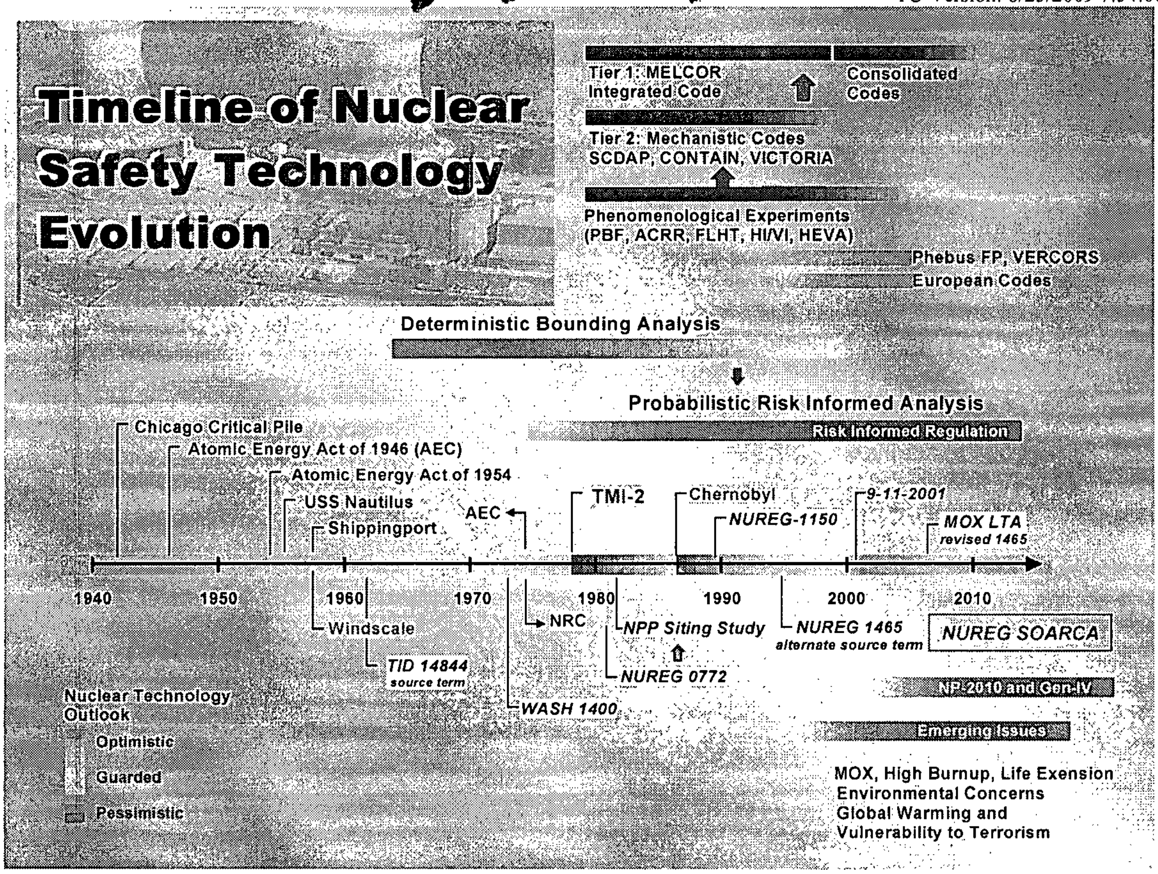


Figure 4 Timeline of Key Nuclear Power Events and Safety Studies.

4.1 Source Term Study Background

One of the earliest estimates of the source term came from the WASH-740 study in 1957 [1]. Three core damage cases were defined with increasing levels of severity. The first case was defined as a situation in which there was a major damage to the core resulting in failure of the vessel. However, the containment remained intact, thus preventing a major release of radioactivity to the environment. This case was subsequently used to define the characteristics of the source term for reactor siting, ~~i.e.~~, TID-14844 [12]. In the other two cases, there were releases offsite.

as developed in

The TID-14844 source term postulated the release of all the noble gases, 50% of the iodine, and 1% of the radioactive solids to the containment. In addition, TID-14844 provided assumptions for containment leakage and for atmospheric transport. However, it was recognized that the procedures and results specified in TID-14844 were approximations, sometimes relatively poor ones, to the results which would be obtained if the effects of the all influencing variables could be recognized and associated with fixed levels of uncertainty – an impossibility in the state-of-the-art at the time [14]. Nevertheless, TID-14844 was codified as "the maximum credible accident" in the siting regulations of 10 CFR Part 100, "Reactor Site Criteria" [15].

more comprehensive that in TID-14844

The next most significant source term study, the Reactor Safety Study (WASH-1400) [3], was the first systematic attempt to provide realistic estimates of public risk from potential accidents in commercial nuclear power plants. The 1975 study included analytical methods for determining both the probabilities and consequences of various accident scenarios. Event trees and fault trees were used to define important accident sequences and to quantify the reliability of engineered safety systems. A ~~more comprehensive~~ list of nine PWR and five BWR source terms was developed. 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. As an improvement over TID-14844, the radioactivity was described using eight chemical categories. The 54 most health-significant isotopes were used in health consequence calculations.

then-

~~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 [16], which 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 was 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% of the likelihood of each release category into the next larger and the next smaller category..

not germane to source term

were of necessity uncertain and reflected a

phenomenological advances in

Subsequently, the NRC documented the technical basis for source terms in NUREG-0772 [17]. 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.e., 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 inclusion of ~~role of~~ containment engineered ~~engineering~~ safety features (i.e. ~~e.g.~~, sprays, suppression pools, and ice condensers). ~~However, much of the quantitative assessment in NUREG-0772 was based on scoping calculations that were only applicable to specific conditions. In particular,~~ The examination of fission product behavior in different regions of the plant with different accidents was conducted in parallel with limited consideration of integral effects. The potential impact of the NUREG-0772 findings on reactor regulation was also examined and the results were documented in NUREG-0771 [18].

NUREG-0771 and NUREG-0772 studies formed the basis for the designation of five accident groups as being representative of the spectrum of potential accident conditions, which were documented in NUREG-0773 [19]. In 1982, the Sandia Siting Study [4] was performed using the NUREG-0773 source terms. ~~The five source terms were assessed to adequately span the range of possible source terms.~~ The source terms were developed from separate effects

B more to next page

isn't this what we are doing now?

sounds pejorative here

2

computer code analyses that were performed in 1978. The source terms were used to calculate accident consequences at 91 United States 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.
the Sandia Siting Study.

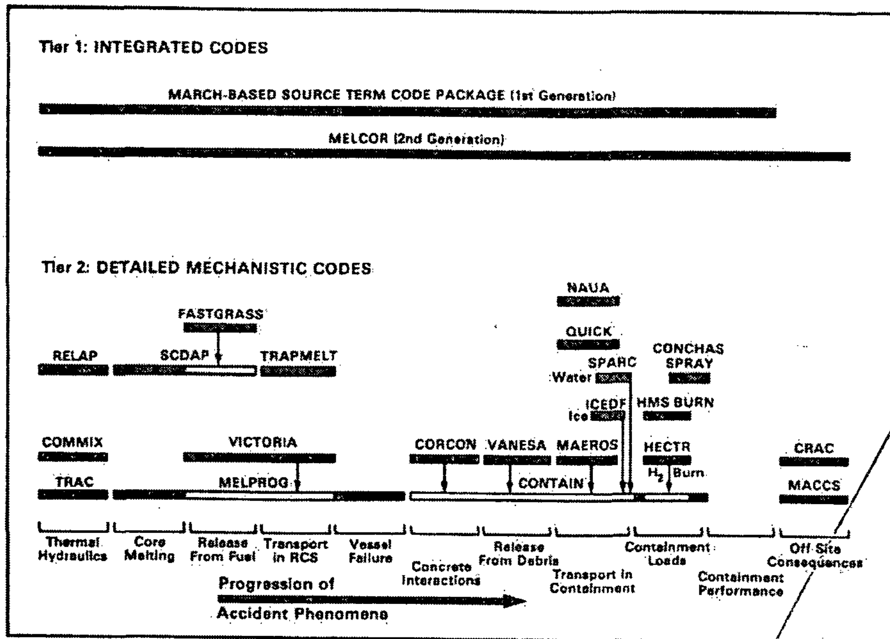
In response to emerging severe accident research technology and computing power, a study was performed at Battelle Columbus Laboratories that involved the development and modification of a number of separate effects severe accident computer codes based on emerging severe accident research. The codes were coupled together to form a code suite that could calculate a complete accident sequence. The source terms for about 25 specific sequences were calculated for five operating plants using the new Source Term Code Package (STCP) code [20]. While the STCP was a significant step forward in deterministic severe accident analysis, the code suite had some significant short-comings. Since 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 was documented in NUREG-0956 [14]. The results from the STCP calculations supported the NUREG-1150 probabilistic risk assessment [5] along with expert judgment and simplified algorithms for sequence-specific source terms.

, containment behavior, and source terms results from

The NUREG-1150 probabilistic risk assessment 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 core damage frequency ~~and risk~~ ~~due to~~ incomplete understanding of reactor systems and severe accident phenomena. The ~~elicitation of~~ expert judgment was used to develop probability distributions for many accident progression, containment loading, structural response, and source term issues. As noted in NUREG-1150, "computer analyses cannot, in general, be used directly and alone to calculate branching probabilities in the accident progression event tree. Since the greatest source of uncertainty is typically associated with the modeling of severe accident phenomena, the results of a single computer run (which uses a specific model) do not characterize the branching uncertainty." It was therefore necessary to use 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, and structural response 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.

that is an alternative to TID-14844.

In 1995, the NRC published NUREG-1465 [13], which defined an ~~alternative~~ accident source term for regulatory applications. ~~The NUREG-1465 source term is considered an alternative to TID-14844, which specified a release of fission products from the core to the reactor containment in the event of a postulated accident involving a "substantial meltdown of the core."~~ NUREG-1465 documents the basis for more realistic estimates of the source term release into containment, in terms of timing, nuclide types, quantities, and chemical form, given a severe core-melt accident. This revised source term is to be applied to the design of future light water reactors (LWRs). Current LWR licensees may voluntarily propose applications based upon it.



Source Term Code Package

Figure 5 Summary of NRC Codes and Two Tier Structure (circa 1986) [14]

4.2 NRC Severe Accident Codes

As a consequence of the need to perform calculations covering a broad range of severe accident phenomena, a two-tier code strategy was developed by NRC in the 1980s (see Figure 5). The STCP was the first Tier 1 integrated analysis code. It was capable of calculating the full scope of the severe accident progression including the radionuclide source term. The STCP was a coupling of ten separate codes that were independently developed to calculate specific aspects of the severe accident progression (e.g., the CORSOR code predicted in-vessel fission product releases and the CORCON code evaluated ex-vessel core-concrete interactions). The Tier 1 codes were originally conceived to include modeling simplifications in order to permit calculation of all phases of the accident. In response to problems associated with coupling many different codes, the MELCOR code development program was initiated to develop a fully integrated code with flexible nodalization capabilities, intrinsic and self-consistent feedback between phenomena, and sensitivity analysis capabilities.

MELCOR was originally conceived as a full-scope Tier 1 analysis tool, then as computer capability increased MELCOR was enhanced to be highly mechanistic in its capability.

The second code tier of severe accident codes that were developed by the NRC was called the detailed mechanistic codes. The detailed mechanistic codes were typically developed and applied in close connection with an experimental program. Their scope was often limited to planning and interpreting experiments. However, the level of detail often far exceeded the comparable models in the Tier 1 codes. Therefore, the mechanistic codes, or the scientific principles within them, ~~are~~ were subsequently used to enhance the integrated codes (i.e., MELCOR). In short, the science of severe accident phenomena is developed in the mechanistic codes and transferred to the integrated codes [14]. MELCOR is discussed in detail in the next section.

Over time, the number of codes maintained by the NRC decreased. MELCOR became the only Tier 1 integrated code and SCDAP/RELAP5, VICTORIA, and CONTAIN were the primary Tier 2 mechanistic codes. SCDAP/RELAP5 calculated the thermal-hydraulic and severe accident response of the primary and secondary systems of the nuclear reactor. It was not used for the radionuclide release and transport calculations. VICTORIA is a detailed code for prediction of the chemical forms of fission products in the primary reactor coolant system. However, it requires thermal-hydraulic boundary conditions from a primary system analysis code. CONTAIN only calculates the containment response and ex-vessel severe accident phenomena.

-ed originally

The MELCOR code was originally conceived as a Tier 1 integrated analysis code for probabilistic risk assessments. In particular, MELCOR models the full-scope of a severe accident including the source term but in a less detailed manner than the detailed mechanistic codes. However, the level of modeling detail in the MELCOR code steadily increased in the 1990s as computer processor speeds increased. The functionality of most of the detailed mechanistic separate effects codes in Figure 5 were fully integrated into MELCOR ~~(e.g., VANESA, CORCON, and SPARC92)~~. Starting in 2000, the NRC initiated a final code consolidation effort to incorporate the SCDAP/RELAP5, VICTORIA, and CONTAIN codes into MELCOR. Once complete, this will provide an increase in efficiency by requiring the maintenance of only one fully integrated code for severe accident analysis (see Figure 6). The scope of the MELCOR code is further discussed in Section 4.3.

Todays

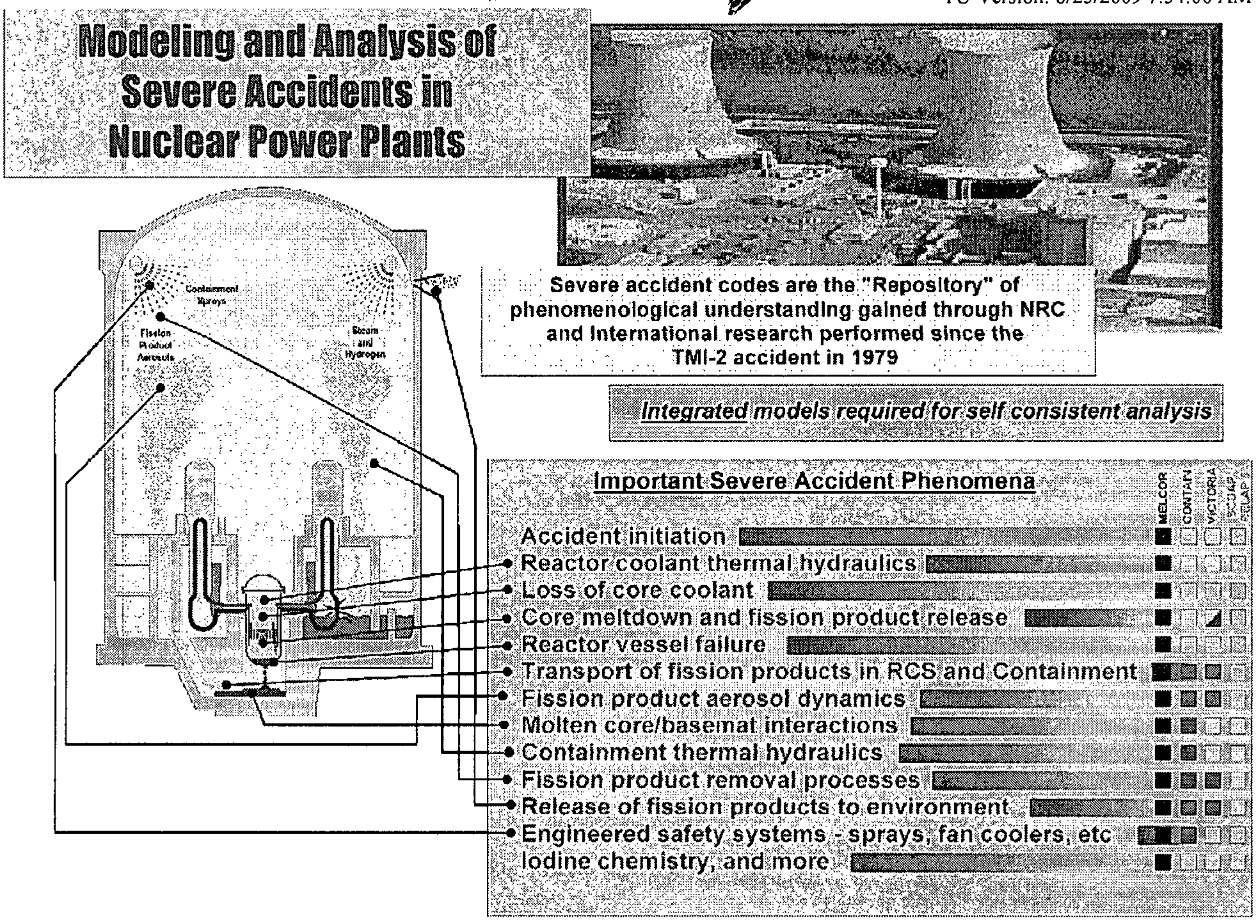


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Figure 6 MELCOR Integration of Separate Effects Codes.

4.3 The MELCOR Code

The MELCOR code is a fully integrated, engineering-level and computer code whose primary purpose is to model the progression of accidents in light water reactor nuclear power plants as well as in non-reactor 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 comprised of three general types of packages: (a) basic physical phenomena (i.e., hydrodynamics (control volume and flow paths), heat and mass transfer to structures, gas combustion, aerosol and vapor physics); (b) reactor-specific phenomena (i.e., decay heat generation, core degradation, ex-vessel phenomena, sprays and engineering safety systems); (c) support functions (thermodynamics, equations of state, other material properties, data handling utilities, equation solvers). As a fully integrated code, MELCOR models all major systems of a reactor plant and their important coupled interactions.

The phenomena modeled are shown schematically in Fig 6, along with the more mechanistic [illegible] codes on which MELCOR is based.

Why is hydrodynamics a basic physical phenomenon and thermodynamics a support function? and EOS and material properties -

The scope of MELCOR includes:

in-vessel
ex-vessel

- thermal-hydraulic response of the primary reactor coolant system, 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~~, sounds as if this is the only source of aerosol - in DA
- 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, and phenomena
- the impact of engineered safety features on thermal-hydraulic and radionuclide behavior.

how's this different for core-concrete

about in-vessel core aerosol generation

Most MELCOR models are mechanistic and the use of parametric models ~~are~~ IS limited to areas of high phenomenological uncertainty where there is no consensus 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. Parameters of this type, as well as such numerical parameters as convergence criteria and iteration limits, are coded in MELCOR as sensitivity coefficients, which may be modified through optional code input. It should be noted that core radioactive nuclide inventories are not utilized by MELCOR rather masses and decay heats of chemical element groups are used. Appropriate code calculations are performed for specific fuel and core design and are carried out to the burnup of interest in order to provide the initial core inventories for MELCOR severe accident analysis (see Section 4.4.1).

of uncertainty and

After the completion of Version 1.8.1 in 1991, the NRC commissioned a peer review using recognized experts from national laboratories, universities, and MELCOR user community [21]. 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 the technical findings of the Committee. The Committee offered a set of major findings that covered the various physics model numerics, missing models, modeling deficiencies, code assessment, and documentation. The findings were incorporated into the NRC research plan that governed the subsequent code development.

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

Major experimental facilities (current facilities are underlined) that provided the bulk of the separate effects and integral effects data used to develop and assess severe accident phenomenology for severe accident codes are delineated in **Table 10** and illustrated in Figures 7 through Figure 9.

too much info

Table 10 Severe Accident Phenomena addressed by U.S. and International Experimental Facilities

Phenomenology	U. S. Facilities	International Facilities
Core degradation	SFD (SNL) PBF (INEEL)	QUENCH (Germany) CORA (Germany) PHEBUS SFD (France)
Hydrogen behavior	Hydrogen testing (BNL)	NUPEC (Japan), RUT (Russia), BMC (Germany)
SA and aerosol behavior	W CSTF westinghouse W 1/7 scale LACE, MACE (EPRI)	ARTIST (Switzerland) PHEBUS-FP (France), ALPHA (Japan)
Fission products – release and transport (including under air-ingress conditions)	HI, VI (ORNL) ACRR ST (SNL)	VERCORS (France) PHEBUS Source Term Separate Effects Program (France)
Direct containment heating	Surtsey, CTTF (SNL) 1/40 scale (Purdue) 1/40 scale (ANL)	FZK DISCO experiments (Germany)
Reactor lower head behavior	OECD OLHF (SNL) CHF downward curved surface (SBLB - Penn State, ULPU – UCSB, CYBL - SNL)	FOREVER (Sweden)
In-Vessel molten core behavior		RASPLAV (Russia), MASCA (Russia), COPO (Finland)
Molten core concrete interaction & melt coolability	OECD MCCI & OECD MCCI Follow-On (ANL)	VULCANO (France)

here the underline means westinghouse but underlined is supposed to mean current facilities -

Phenomenology	U. S. Facilities	International Facilities
Fuel coolant interaction	WFCI (Univ. of Wisconsin) COREXIT (ANL)	KROTOS & FARO (JRC) ALPHA (Japan) TROI (Korea) KROTOS (France), MFMI for ACR-700 (Canada)
Cladding under air oxidation	Cladding oxidation (ANL)	Cladding oxidation (France – planned)
Iodine	ORNL ACE (EPRI)	PHEBUS-FP & PHEBUS-Source Term Separate Effects Test Program (France), FALCON (U.K.)

Most of the U.S. experiments were carried out in the late 1980s and 1990s, and most of the U.S. experimental facilities have been de-activated. However, through the NRC Cooperative Severe Accident Research Program (CSARP) or bi-lateral agreements, NRC has access to international severe accident research. Details on some of the on-going severe accident research programs that NRC is participating under bi-lateral agreements are described below in **Table 11**.

The NRC and others invested heavily in the experimental and analytical characterization of severe reactor accidents ~~that dominate the risk to the public posed by the use of nuclear reactors to produce electrical energy~~. A substantial technology was established to understand the progression of reactor accidents and the radiological consequences of such accidents. Once the objectives of the program were met, the programs were concluded. Two key objectives of the NRC severe accident research were (1) to assess whether the phenomena were sufficiently understood to estimate risks to the level of confidence needed, and (2) to provide assurance of adequate protection.

Experimental research on severe accidents is continuing in other countries to examine or independently confirm uncertain and complex severe accident phenomena. ~~Furthermore, regulatory practices and siting constraints in other countries led to more restrictive acceptance criteria than 'adequate protection.'~~ Many ~~substantial experimental and code development programs are under way around the world~~. The NRC maintains efforts to update their severe accident analysis capabilities with research results from international programs. The body of knowledge developed from the NRC's past work and ongoing work are ~~systematized in useable form~~ in the MELCOR accident analysis code. ~~At the same time, the NRC participates in international cooperative research programs to obtain data for validating the MELCOR code and improving its accuracy and realism. Current agreements include the following:~~

ours isn't good enough

incorporated

MASCA

MASCA is an experimental study under way in Russia on the behavior of reactor core debris in the lower plenum of a reactor pressure vessel. Results of these studies provide data on debris coolability in the lower plenum.

ARTIST

ARTIST is an experimental study under way in Switzerland to measure the aerosol removal on the secondary sides of steam generators during accidents at PWRs that bypass reactor containments. Such bypass accidents are often risk dominant for PWRs. The high risks associated with such accidents may stem from conservatism in the aerosol decontamination assumed in accident analysis models for steam generators. Test results are expected to provide the basis for more realistic analyses of these accidents, such as the SOARCA scenarios with tube failure.

PHEBUS-FP

The PHEBUS-FP program consists of five large-scale, in-pile integrated tests of fuel degradation, fission product release, radionuclide transport through the reactor coolant system, and aerosol behavior in the containment. These tests have been designed to validate reactor accident models. Results for code validation are being produced by this international program. Additional information is being provided by supporting separate-effects experimental programs, such as the French program VERCORS, to investigate fission product release from fuels under accident conditions.

OECD-MCCI

OECD-MCCI is an experimental study of the viability of using an overlying layer of water on reactor core debris that has escaped the reactor coolant system and is interacting with structural concrete of reactor containments. This research will provide data for improved or new models of core debris coolability for accident analysis codes.

These international programs are providing the bases for validating MELCOR. The MELCOR code has been used to help resolve regulatory issues such as the need for hydrogen igniters in ice condenser and Mark III containments and risk-informing 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors" [22]. MELCOR analyses are also important for the certification of the AP1000 and ESBWR advanced LWR designs. MELCOR analyses will likely be instrumental in the certification of other advanced reactor designs.

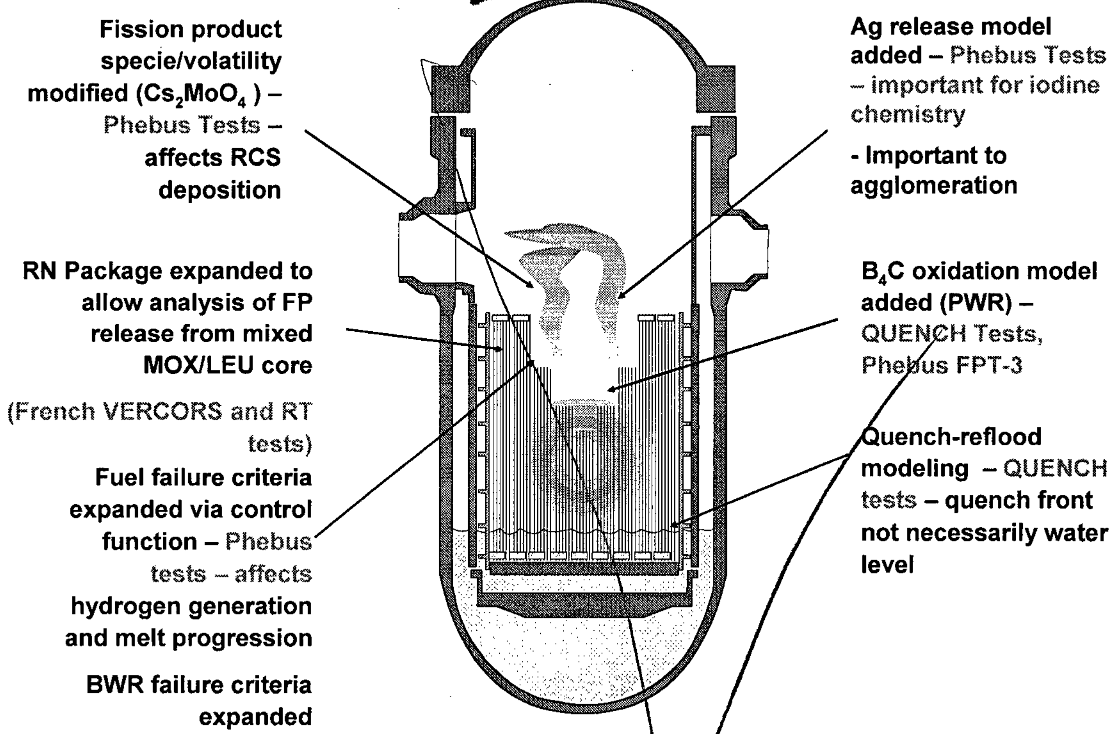

Figure 7 MELCOR Development Activities for Core Heatup and Fission Product Release.

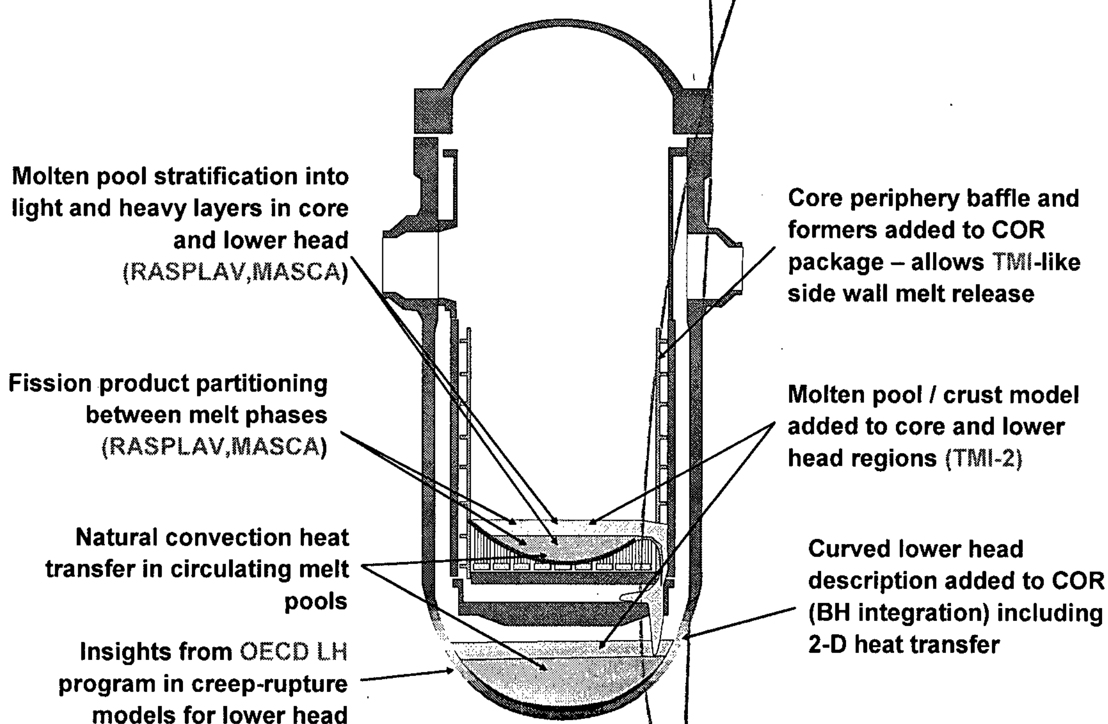

Figure 8 MELCOR Development Activities for Late Phase Melt Progression.

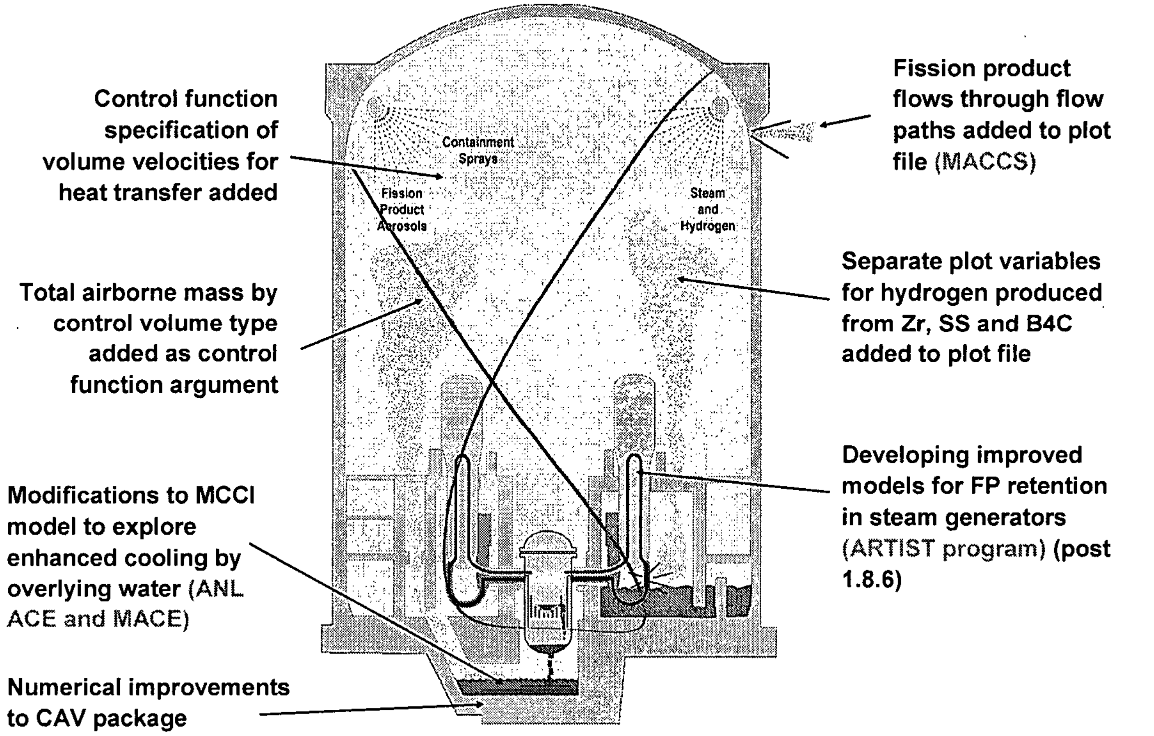

Figure 9 MELCOR Development Activities for Containment Modeling.

Table 11 PHEBUS-FP/PHEBUS Source Term Follow-On, ARTIST, OECD-MCCI and OECD MASCA programs

Facility/Country	Major products	Usage
PHEBUS-FP and PHEBUS Source Term Separate Effects Test Program Institut de Radioprotection et de Sûreté Nucléaire (IRSN), France	Fission product releases and degradation of UO_2 fuel (including burnup >40 GWD/Mt) and MOX fuel under severe accident conditions, and the effects of air ingress on core degradation and fission product release	Validate the NUREG-1465 source term MELCOR code assessment - core degradation - fission products release - iodine behavior - fission products chemistry - air-ingress impact on FP release Revised NUREG-1465 for MOX and high burnup fuel
ARTIST - Paul Scherrer Institute, Switzerland	Experimentally determine the potential mitigation of radioactive material releases through secondary side of a steam generator	Improve source term bypass models

Facility/Country	Major products	Usage
OECD MCCI program - Argonne National Laboratory (USA) - current and follow-on program	Separate effects experiments to further addressed the ex-vessel debris coolability issue	Assess/develop coolability models in MELCOR
OECD MASCA program - Russian Research Center, Kurchatov Institute (Russia) Current and follow-on program	Separate experiments to provide data on the effect of (a) chemical behavior and interactions of the molten core debris (UO_2, ZrO_2-Zr) and reactor structural materials (steel, boron carbide); and (b) partition of fission products between oxidic and metallic layers in the stratified molten pool, and the resulting heat load distribution in the lower head of the reactor pressure vessel (RPV). Similar materials behavior for new reactors.	Molten core debris behavior both in-core and in the lower head of a RPV, which in turn address the question on lower head integrity (i.e., the in-vessel retention of molten core debris) under severe accident conditions. (Assess Severe Accident Management Strategy)

4.4 MELCOR Modeling Approach

A high-level description of the MELCOR models that were used for the SOARCA project is presented in Section 4.4.1. 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. More detailed information describing the plant models is provided in the plant-specific analysis reports (i.e., Volumes ~~III~~ II and ~~IV~~ III for Peach Bottom and Surry, respectively).

The progression of events in a severe accident contains uncertainties. The procedure to define the best practices approach to modeling important and uncertain phenomena is described in Section 4.4.2. ~~Volume II provides a more detailed description of the best practices modeling approach.~~

Section 4.4.3 summarizes some recent changes to the radionuclide release and cesium speciation modeling, which is important to the source term results. Finally, the methodology to calculate the radionuclide inventory is described in Section 4.4.4.

4.4.1 Plant Models

The MELCOR models used in the SOARCA source term calculations represented the state-of-the-art. As part of the SOARCA program, the MELCOR models were updated to the most recent version of the MELCOR code.[3] The scope of the models included:

[3] MELCOR Version 2.0 was released during the initial phase of the SOARCA program. Version 2.0 is based on identical physics models as Version 1.8.6 but has been modernized to use FORTRAN 90 and a new input format.

the side of the

- Detailed 5-ring reactor vessel models
- Representation of the primary reactor coolant systems (and secondary steam generator through the main steam isolation valve for Surry)
- Representation of the primary containment
- Representation of the Peach Bottom reactor building and the Surry auxiliary building, which w~~h~~ere radionuclide pathways in some scenarios
- Representation of the emergency core-cooling systems (and the auxiliary feedwater system for Surry)
- Representations of the emergency, portable water injection systems

Through the best practices updates to each deck, the following new models were specified for both plants for these important but uncertain phenomena or equipment responses.

- Safety relief valve failure models for normal or high temperature conditions,
- An additional thermo-mechanical fuel collapse model for heavily oxidized fuel following ~~molten zircaloy breakout~~, we got U_3O_8 ? or ZrO_2 ?
- Enhanced lower plenum coolant debris heat transfer that recognizes break-up and multi-dimensional cooling effects not present in the one-dimensional counter-current flooding model in older versions of MELCOR (e.g., [23]),
- Updated, plant-specific chemical element masses and decay heats (see Section 4.4.4),
- A new ORNL-Booth chemical element release model and new Cs speciation model (see Section 4.4.3),
- Vessel failure based on gross failure[4] [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~~, and
- Enhanced ex-vessel core debris heat transfer that recognizes multi-dimensional effects and rates measured in MACE tests [25].

A summary of recent enhancements to the MELCOR Peach Bottom and Surry models for the SOARCA program are presented in Section 4.4.1.1 and 4.4.1.2, respectively.

4.4.1.1 Peach Bottom MELCOR Model

The Peach Bottom MELCOR model was originally developed for code version 1.8.0 at Brookhaven National Laboratories. The model was subsequently adopted ~~by J. Carbajo~~ at Oak Ridge National Laboratories to study differences in fission product source term behavior predicted by MELCOR 1.8.1 and those generated for use in NUREG-1150 using the Source Term Code Package (STCP) [26]. Starting in 2001, Sandia National Laboratories made considerable refinements to the BWR/4 core nodalization to support the developmental

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[4] A more complete discussion of this model is presented in ~~Volume II and~~ the MELCOR manual [15]. A penetration failure model was not used, because the timing differences between gross lower head failure and penetration with and with the available penetration model is not significant to the overall accident progression (i.e., minutes difference). Also, Sandia Lower Head Failure (LHF) tests showed gross creep rupture of the lower head was measured to be the most likely mechanism for vessel failure [24].

assessment and release of MELCOR 1.8.5. These refinements concentrated on the spatial nodalization of the reactor core (both in terms of fuel/structural material and hydrodynamic volumes) used to calculate in-vessel melt progression.

Subsequent work in support of several U.S. 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 a wider range of severe accident sequences. These enhancements include:

control rod drive ? Hydraulic System

- modifications of modeling features needed to achieve steady-state reactor conditions (recirculation loops, jet pumps, steam separators, steam dryers, feedwater flow, CRDHS, main steam lines, turbine/hotwell, core power profile),
- new models and control logic to represent coolant injection systems (RCIC, HPCI, RHR, LPCS) and supporting water resources (e.g., CST with switchover), and

automatic depressurization System

- new models to simulate reactor vessel pressure management (safety relief valves, safety valves, ADS, 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 primary and secondary containment. The drywell portion of primary containment has been sub-divided 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 (i.e., secondary containment). The third area has focused on bringing the model up to current "best practice" standards for MELCOR 1.8.6 (see Section 4.4.2). The fourth area of model improvements included a new radionuclide inventory and decay heat based on the recent plant operating history (see Section 4.4.4).

While not new for SOARCA, the MELCOR Peach Bottom model includes a multi-region 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, the liner will fail if the debris temperature is above the carbon steel melting temperature.

The MELCOR Peach Bottom model is more fully described in Volume ~~III~~ II. The approach to modeling important uncertain phenomena is briefly described in Section 4.4.2 ~~and more fully described in Volume II~~. The MELCOR nodalization diagrams for Peach Bottom are shown in Figure 10.

4.4.1.2 Surry MELCOR Model

The Surry MELCOR model applied in this report was originally generated at Idaho National Engineering Laboratories (INEL) in 1988. The model was periodically updated by Sandia

National Laboratories (1990 to present) for the purposes of testing new models, advancing the state-of-the-art in modeling of PWR accident progression, and providing support to decision-makers at the ~~U.S. Nuclear Regulatory Commission (~~NRC~~)~~ for analyses of various issues that may affect operational safety. Significant changes were made during the last twenty years 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 reactor coolant nodalization were updated using the SCDAP/RELAP5 Surry model to include a five ring vessel nodalization and counter-current hot leg representation for natural circulation flow [27]. The current MELCOR Surry model is a culmination of these efforts and represents the state-of-the-art in modeling of potential PWR severe accidents.

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 MELCOR Version 1.8.6 core modeling. These enhancements include:

what is the 3rd only 2 described

- 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 both in 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, and
- a control rod silver aerosol release model, ~~and~~
- ~~addition of the new ONRL-Booth radionuclide release model for modern high-burn-up 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 capabilities of MELCOR to a wider range of severe accident sequences. These enhancements included:

- update of the containment leakage model,
- update of core degradation modeling practices, what besides all the stuff already mentioned
- modeling of individual primary and secondary 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 and turbine-driven auxiliary feedwater models with control logic for plant automatic and operator cooldown responses,
- new turbine-driven auxiliary feedwater models for steam flow, flooding failure, and performance degradation at low pressure,
- nitrogen discharge model for accumulators,

does all this belong in section 4.4.1 rather than Surry specific

- update of the fission product inventory, the axial and radial peaking factors, and an extensive fission product tracking control system, and
- improvements to the natural circulation in the hot leg and steam generator and the potential for creep rupture.

The MELCOR Surry model is more fully described in Volume III ~~IV~~. The approach to modeling uncertain phenomena is briefly described in Section 4.4.2 ~~and more fully in Volume II~~. The MELCOR nodalization diagrams for Surry are shown in Figure 11.

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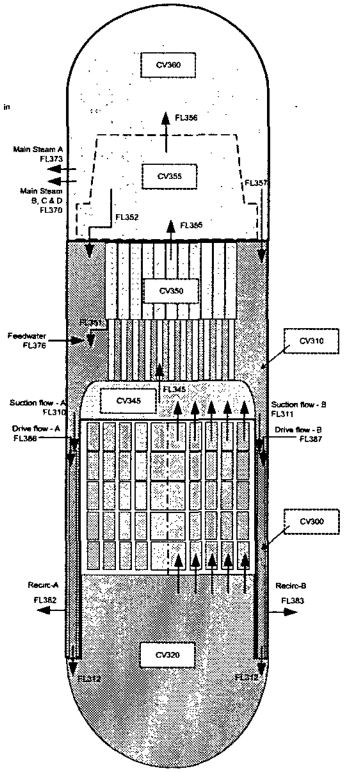
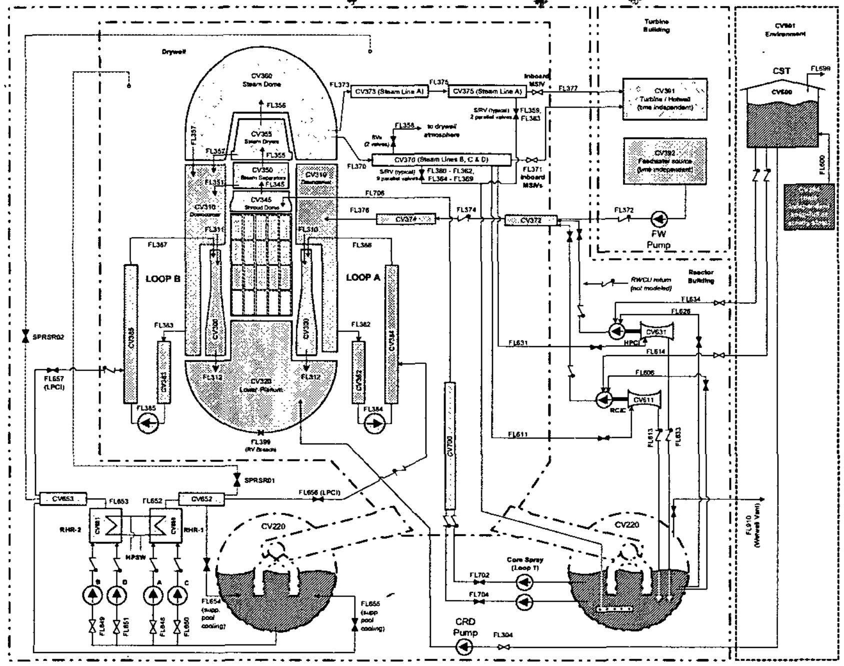
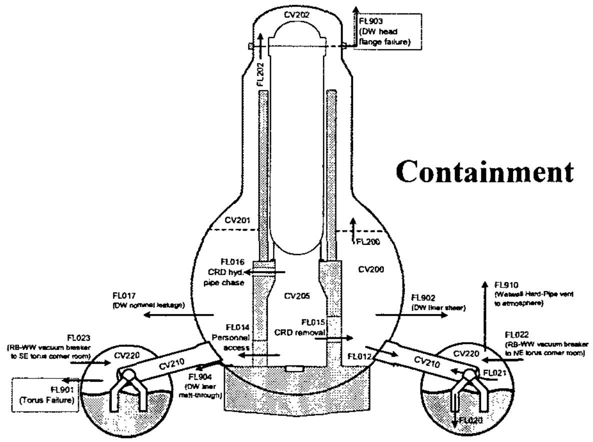
Reactor Vessel



Reactor coolant system and ECCS



Containment



Reactor Building

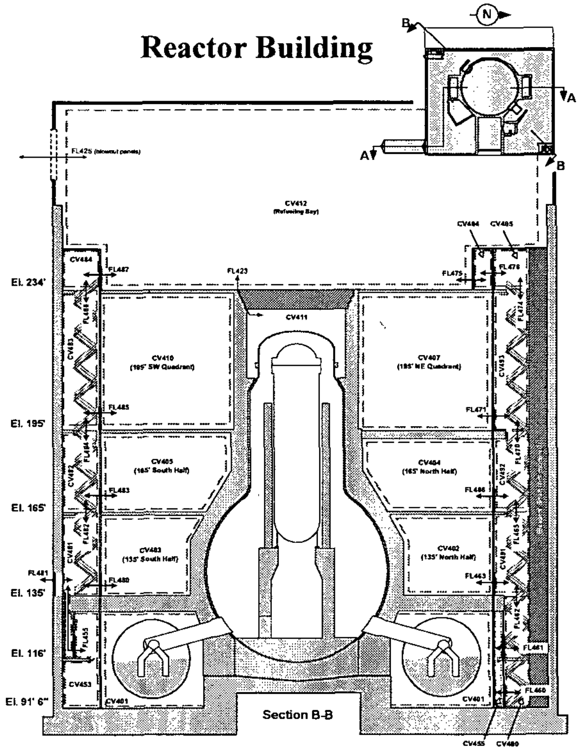


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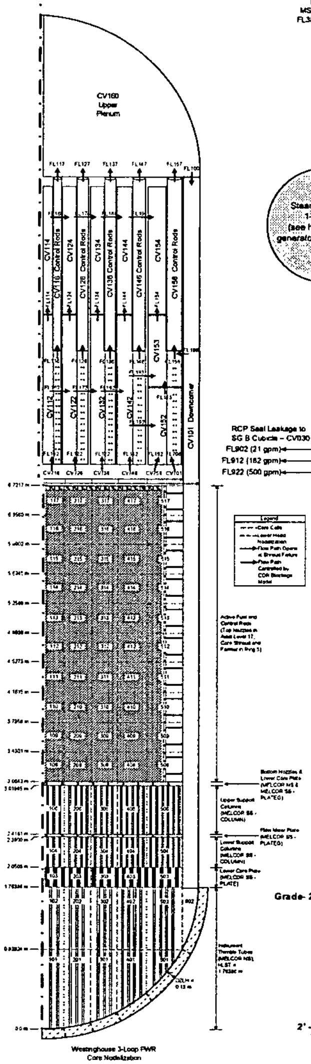
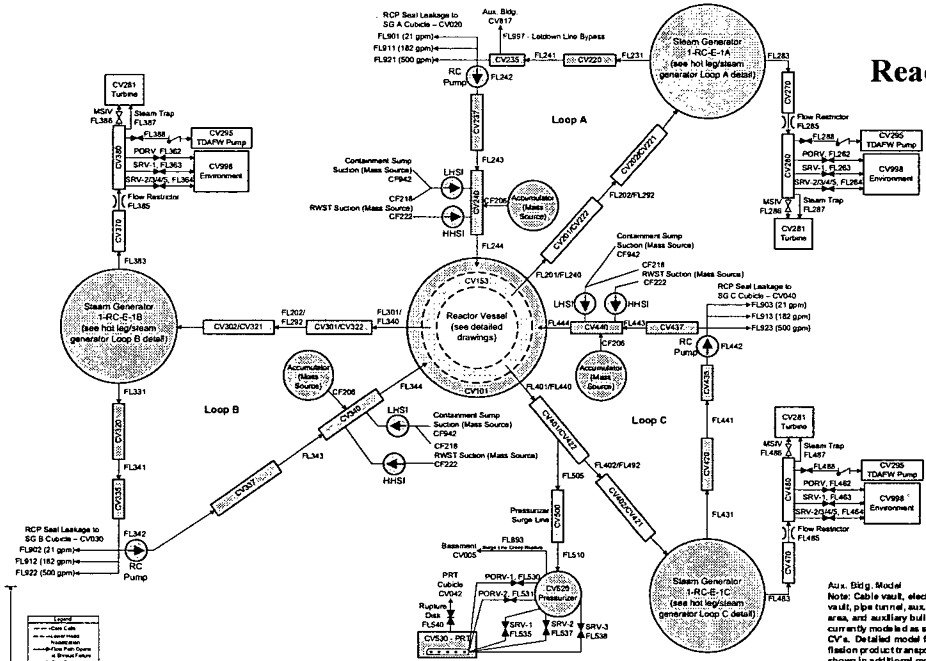
Figure 10 The Peach Bottom MELCOR Nodalization.

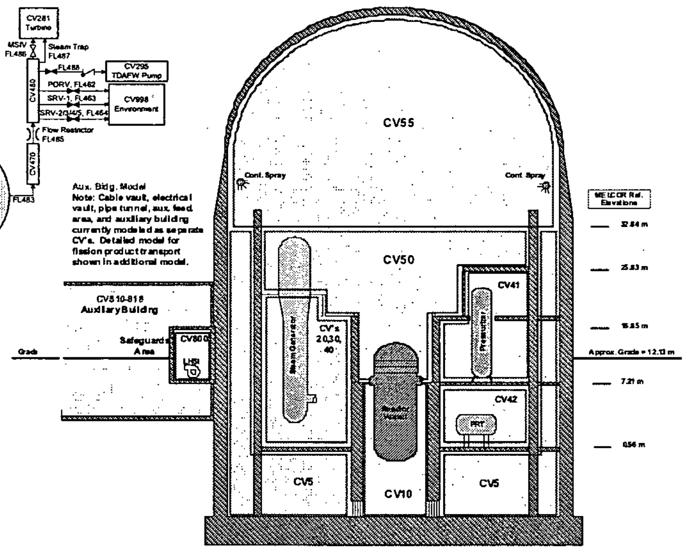
Reactor Vessel



Reactor coolant system



Containment



Auxiliary Building

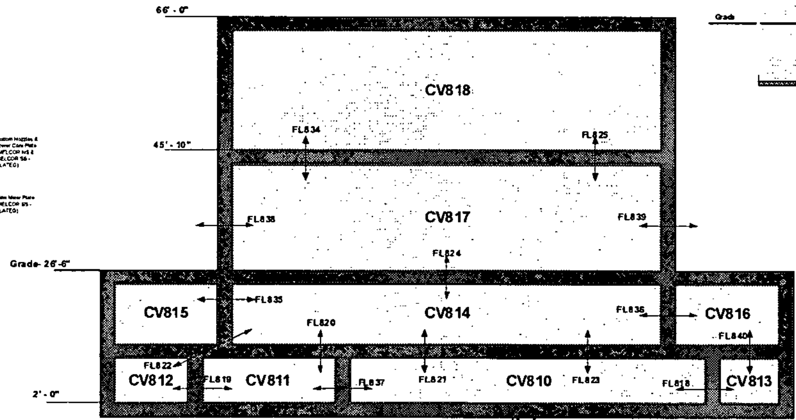


Figure 11 The Surry MELCOR Nodalization.

other plot mostly

4.4.2 ~~Best Modeling Practices~~

The accident progression analysts developed a list of key uncertain phenomena that can have a significant effect on the progression of the accident. Each issue was outlined and a recommended modeling approach or base case values were identified in plant-specific reports for Peach Bottom (Volume ~~III~~ II) and Surry (Volume ~~IV~~ III). ~~A discussion of the specific modeling practices are described in Volume II.~~

An independent expert panel was assembled to review the proposed approach. The review was conducted during a public meeting sponsored by the NRC on August 21–22, 2006 in Albuquerque, New Mexico. The expert panel review examined the best modeling practices for the application of the severe nuclear reactor accident analysis code MELCOR for realistic evaluation of accident progression, source term, ~~and offsite consequences~~. The panel also reviewed a set of code enhancements and as well as considered ~~consideration~~ of the SOARCA project in general. The needs definition.

For operator actions, a sensitivity calculation was performed for each accident sequence to quantify the impact of the operator response.

A second approach to address uncertainties will be performed during a separate analysis task to be conducted subsequent to this initial analysis. In that task, the importance and impact of alternative settings or approaches for uncertainty will be evaluated with respect to a specific scenario.

Several early containment failure modes have been excluded from ~~the~~ SOARCA ~~due~~ because of their assessed low-likelihood. Hence, they were not considered in the best practices MELCOR modeling. These include;

1. Alpha mode containment failure, which is an in-vessel steam explosion during melt relocation that simultaneously fails the vessel and the containment. A group of leading 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, posing little or no significance to the overall risk from a nuclear power plant.

2. Direct containment heating (DCH) causing containment failure in PWR containments. Decades of NRC research show an early failure of reactor coolant system due to high temperature natural circulation will depressurize the system prior to vessel failure. In the unlikely event there is a high-pressure vessel failure, the resolution of the DCH issue found ~~the early~~ containment failure to be very unlikely [29]. from this phenomenon

3. Early containment failure due to drywell liner melt-through in wet cavity in Mark I containments (e.g., Peach Bottom). Through a detailed assessment of the issue, it was

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].

4.4.3 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~~ comparison with ~~to~~ recent radionuclide release tests. These assessments identified an alternative set of Booth diffusion parameters recommended by ORNL (ORNL-Booth) [32], which produced significantly improved release signatures for Cs and other fission product groups. Some adjustments to the scaling factors in the ORNL-Booth model were made for selected fission product groups, including UO_2, Mo and Ru in order to gain better comparisons with the 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 French VERCORS tests [34], and the comparisons ~~was~~ were as favorable or better than the original CORSOR-M MELCOR default release model. These modified ORNL-Booth parameters were implemented into the MELCOR code ~~as new defaults~~ for the SOARCA project. as

While significant improvements in release behavior were obtained for the analysis of the FPT-1 test with the ORNL-Booth parameters, 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 700C, explaining deposits in the hot upper plenum of the Phebus test section, and deposition patterns in the cooler steam generator tubes. ~~In recognition of response~~ Therefore, a Cs_2MoO_4 radionuclide class was defined with the vapor pressure of Cs_2MoO_4 and the release coefficients developed for Cs. While having little effect on the net release of Cs, this change had a significant effect on the release of Mo. The Mo vapor pressure is so exceedingly low that the net release is limited by the vapor pressure transport term. Since 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.

The radionuclide input was reconfigured to (a) represent the dominant form of Cs as Cs_2MoO_4, (b) represent the dominant form of I as CsI, and (c) represent the gap inventories consistent with the NUREG-1465 recommendations [13]. The MELCOR radionuclide transport, deposition, condensation/evaporation, and scrubbing models were all ~~activated~~ used. The model for chemisorption of Cs to stainless steel was activated. In addition, the hygroscopic coupling between the steam/fog condensation/evaporation thermal-hydraulic solutions to the airborne aerosol size and mass was also ~~activated~~ used [31].

4.4.4 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 radio-isotopes in a nuclear

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neutron flux

The BLEND3 code was developed from previous work performed by Oak Ridge National Laboratory 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 radio-isotopic inventories. TRITON uses ~~the~~ generic fuel assembly data and ties it 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 radio-isotopic inventory from the relevant ORIGEN output files. Finally, using the input volume fractions for the three different cycles of fuel, it 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 which allows the user to automate the execution of a series of SCALE calculations.

4.4.4.3 Peach Bottom ~~Model~~

The Peach Bottom model is based on the Global Nuclear Fuel (GNF) 10x10 (GE-14C) fuel assembly. The GNF 10x10 is representative of a limiting fuel type actually being used in commercial BWRs. The GEH 10x10 model is illustrated in Figure 12. The model is very detailed for this application. The only significant assumption was that the part length rod portion of the reactor was modeled as a full assembly.

Twenty-seven different TRITON runs were performed to model three different cycles of fuel at nine different specific power histories. The specific power histories ranged from 2 MW/MTU to 45 MW/MTU to cover all expected BWR operational conditions. For times before the cycle of interest, an average specific power of 25.5 MW/MTU was used. For example, for second cycle fuel, the fuel was burned for its first cycle using 25.5 MW/MTU, allowed to decay for an assumed 30 day refueling outage and then 9 different TRITON calculations were performed with specific powers ranging from 2 to 45 MW/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~~.

as a function of time after shutdown.

calculate the decay of the radionuclides in

mass of each chemical element group

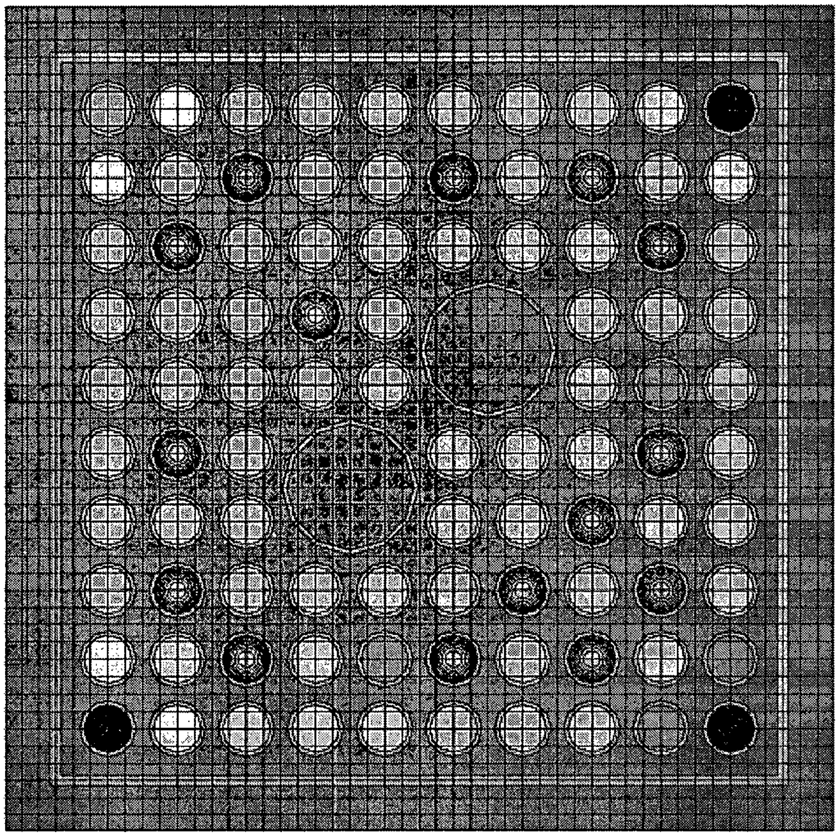

Figure 12 Schematic of Modeling Detail for BWR GNF 10x10 Assembly.

call KC

4.4.4.4 Surry ~~Model~~

Previously, detailed input was developed for Surry in a separate NRC program on the source term from high-burnup uranium (HBU) fuel at the end of the fuel cycle. It used the same methodology as Peach Bottom (Section 4.4.4.3). The actual mid-cycle decay power is lower. However, the SOARCA schedule did not allow for a current operation, decay heat evaluation as was done for Peach Bottom. This is important/not important?

than the PB calc?, The actual Surry midcycle?

4.4.4.5 Evaluation of the Results

There are very few measurements of decay heat in existence and those that do exist 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. The best known source of decay heat predictions is summarized in Regulatory Guide 3.54 and results from the guide will be used to assess the predictions in the current study [37]. 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 approximately 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 radio-isotopic predictions presented in this study would be of limited use given the cycle specific nature of this work. However, it is of benefit 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 Germina Ilas and Ian C. Gauld [40]. These assessment reports use data from Calvert Cliffs, Obrigheim, San Onofre, and Trino Vercelles PWRs. The third report summarized comparisons to decay heat measurements from 4 different BWR assemblies. The information in these reports demonstrates that TRITON predicts fission product and actinide inventories at a level of accuracy consistent with other methods.

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5.0 OFF-SITE CONSEQUENCE MACCS2 ANALYS~~ES~~IS ~~SUMMARY~~

MACCS2 [41] has been developed by Sandia National Laboratories for the NRC over the past decade. It is a consequence analysis code for evaluating the impacts of atmospheric releases of radioactive aerosols and vapors on human health and on the environment. It includes all of the relevant dose pathways: cloudshine, inhalation, groundshine, and ingestion. Because it is primarily a probabilistic risk assessment tool, it accounts for the uncertainty in weather that is inherent to an accident that could occur at any point in the future.

In 2001, the NRC initiated an effort to create a Windows-based interface and framework for performing consequence analyses. This effort was intended to address the following needs:

- To simplify and make more intuitive the effort required to create or modify input files,
- To reduce the likelihood of user errors in performing consequence analyses,
- To enable the user to simply and conveniently account for uncertainties in most of the real-valued input parameters, and
- To displace the original batch framework with a Windows-based framework.

The result of this development effort is the WinMACCS code. WinMACCS is currently integrated with an updated version of MACCS2, COMIDA2, and LHS (Latin Hypercube Sampling) to perform all of the required ~~functionality~~. calculations and data handling.

The version of MACCS2 used for SOARCA is 2.4.0.1. 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.,
- Capability to describe wind directions in 64 compass directions (instead of 16);
- Increases in limits on several input parameters, e.g., a limit of 200 plume segments instead of the old limit of 4; and
- Up to 20 emergency-phase cohorts (instead of the original limit of 3) to describe variations in emergency response by segments of the population;
- Enhancements in treatment of evacuation speeds to account for road type and precipitation conditions;
- Capability to run on a cluster of computers instead of an individual processor;
- Addition of several options for dose response.

Some of this development has been undertaken specifically to support the SOARCA work.

Specific aspects of the consequence modeling in SOARCA that depart from previous studies, such as NURG-1150 [5], are described in the subsequent subsections.

5.1 Weather Sampling

The weather-sampling strategy adopted for SOARCA uses the non-uniform weather binning approach ~~in WinMACCS~~. This approach 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 wind speed, stability class, and the occurrence of precipitation. This sampling strategy was chosen as a means of improving the statistical representation of the weather. This point is discussed further in the subsequent paragraphs.

The weather bins are defined in a standard way that has origins in the NUREG-1150 [5] analyses. A set of 16 weather bins differentiate 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 km (20 mi). 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 [5] and documented in the MACCS2 User's Manual [41].

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 1000 weather trials for both Peach Bottom and Surry. The strategy also results in each weather trial having a weight that is used in averaging the results. The weight reflects the number of weather samples in the bin and the number of bin samples chosen.

Previous calculations, such as NUREG-1150, used about 125 weather trials but also used 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 x 16 = 1750 results for the computational price of 125. although These results were not fully independent.

MACCS2 does not allow the rotation option to be used in concert with the network evacuation option (described in Section 5.2), and so rotation could not be used for SOARCA. The strategy adopted for SOARCA was chosen as a compromise between obtaining adequate statistical significance and of keeping central processing unit (CPU) time at a reasonable level.

5.2 Weather Data

Meteorological data used in the SOARCA project consisted of a year of hourly meteorological data ~~for each licensed site participant~~ (8,760 data points per site for each meteorological parameter). This was primarily accomplished via a cooperative effort with the licensee. As a comparative tool, site-specific latitudes and longitudes (or available locations closest to the site) were used to collect wind speed (in meters per second or m/s), wind direction (in degrees), precipitation (in 100th inches), and stability (defined as $\Delta T/\Delta P$) data from the National Climatic Data Center (NCDC) database. The meteorological data parameters were formatted for the

how good was the comparison?

MACCS2 (MELCOR Accident Consequence Code System, version 2) computer code.

NRC staff performed quality assurance evaluations of all meteorological data presented using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data" [42]. Further review was performed using computer spreadsheets. NRC staff ensured there was joint data recovery rate in the 90th percentile, which is in accordance with Regulatory Guide 1.23 [43] for the wind speed, wind direction, and atmospheric stability parameters. Additionally, atmospheric stability was evaluated 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 were retrieved from the EPA SCRAM database[5] (using years 1984-1992). Data needed for MACCS2 includes 10-meter wind speed, 10-meter wind direction in 64 compass directions, stability class (via Pasquill-Gifford scale and using representative values of 1-6 for stability classes A-F/G), hourly precipitation, and diurnal (morning and afternoon) seasonal mixing heights.

Where did we get precip data?

5.2.1 Summary of Weather Data

of the two years examined

A summary of the meteorological statistical data is presented in **Table 12**, which shows that the predominant ground-level wind directions were generally *blowing to* the same direction during each ~~annual period~~ for each nuclear site. It also shows that the annual average wind speeds were generally low, ranging from 2.02 to 2.63 m/s at ground-level. The atmospheric stability frequencies were found to be 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. The wind direction and atmospheric stability (unstable, neutral, and stable) data are shown in Figure 13 through Figure 14 for the years that were actually used in the consequence analyses, which were 2005 for Peach Bottom and 2004 for Surry.

Table 12 Statistical Summary of Raw Meteorological Data for SOARCA Nuclear Sites

	Peach Bottom	Peach Bottom	Surry	Surry
	Year 2005	Year 2006	Year 2001	Year 2004
Avg. Wind Speed (m/s)	2.25	2.63	2.02	2.28
Predominant Wind	SSE	N	NE	NNE
No. Hours w/Precipt	588	593	388	521
% Unstable Stability	21.43	20.56	7.09	3.94
% Neutral Stability	63.97	62.34	69.67	77.59
% Stable Stability	14.60	17.10	23.24	18.47
Joint Data Recovery	97.53%	99.25%	99.58%	99.24%

Note: Year 2004, as used in the Surry meteorological analysis, is a leap year (8784 total hourly data points versus 8760 hourly data points for a regular annual period). The "Predominant Wind" indicates the direction to which the wind blew most frequently. *Precipt* is short for precipitation. *Stability* is short for Atmospheric Stability.

[5] EPA SCRAM website: http://www.epa.gov/scram001/mixingheightdata.htm

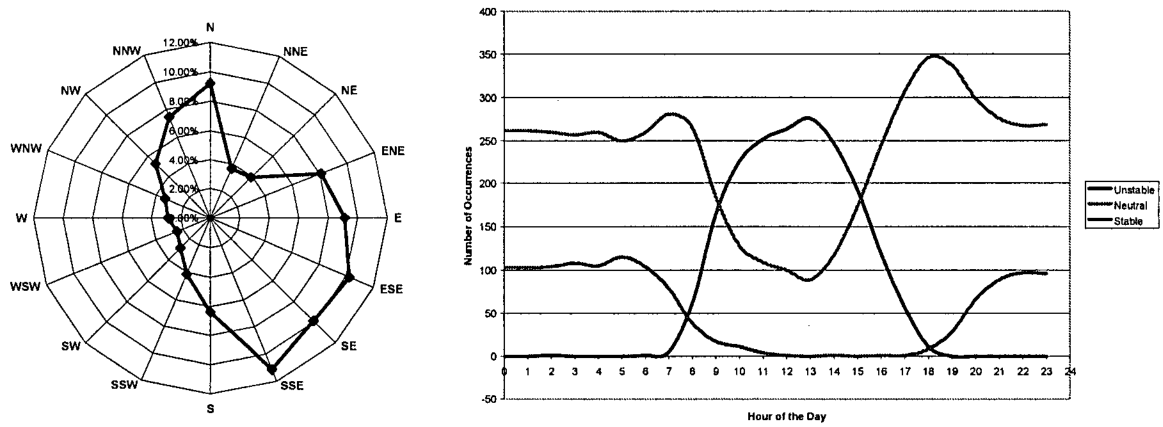

Figure 13 Peach Bottom – Year 2005 – Wind Rose and Atmospheric Stability Chart

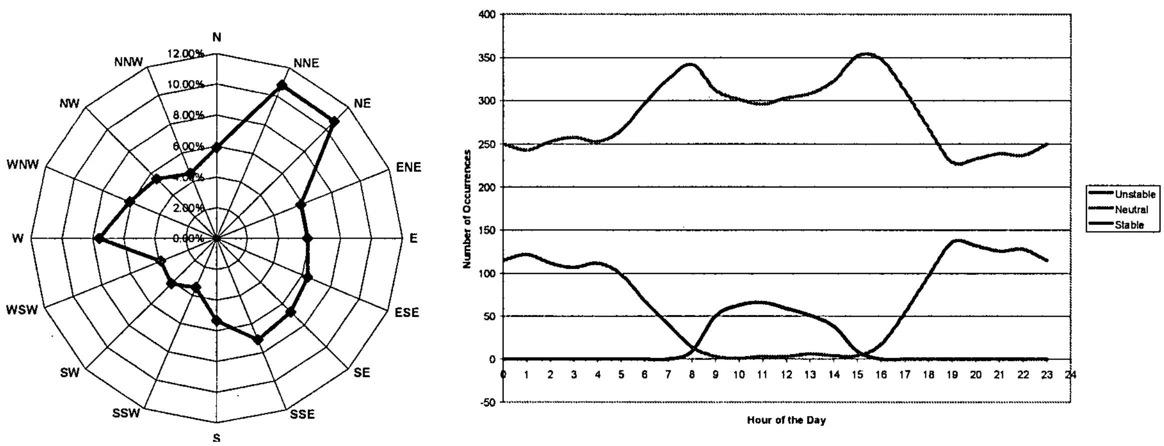

Figure 14 Surry – Year 2004 – Wind Rose and Atmospheric Stability Chart

5.3 Emergency Response Modeling

An objective of the SOARCA project was to model emergency response in a realistic and practical manner based on site-specific emergency planning documentation. Emergency response programs for nuclear power plants (NPPs) are designed to protect public health and safety in the event of a radiological accident. These emergency response programs are developed, tested, and evaluated and are in place as an element of defense in depth in the unlikely event of an accident. Integrating the response plan elements and a best estimate of the protective actions that would be taken by the public was undertaken to improve the overall fidelity of the consequence analyses.

for the SOARCA project

Detailed emergency response planning is in place within the 10-mile EPZ with consideration that such planning provides a substantial base for expansion of response efforts in the event that this proves necessary [44]. The actions taken by offsite response organizations (OROs) are designed to reduce risk to the public in the unlikely event of an accident and these actions would be implemented in the case of an emergency. Site specific information was obtained from OROs to support development of timelines by which protective actions would most likely be implemented, including early actions such as evacuation of schools following declaration of a site area emergency.

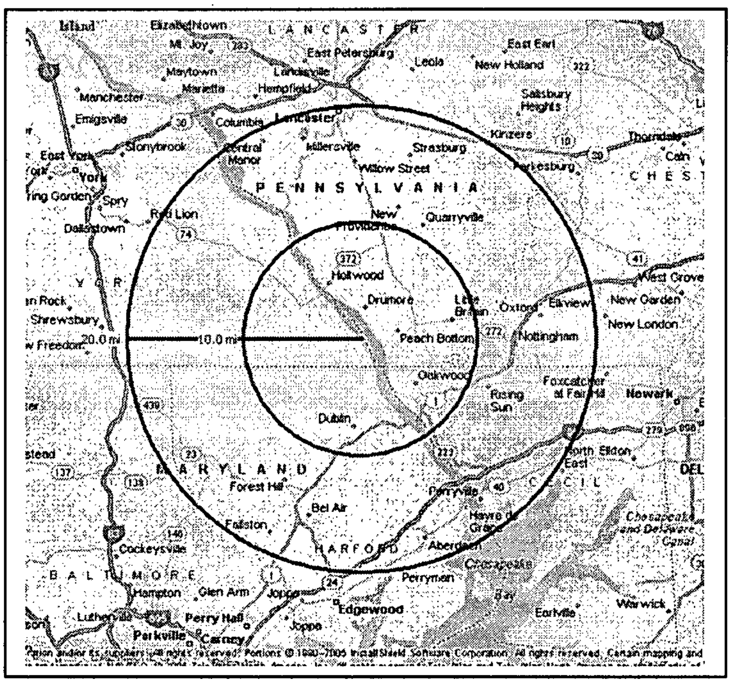

Figure 15 Emergency Preparedness Protective Action Boundaries for the Peach Bottom Site

The SOARCA project provided an opportunity to assess response by populations within the 10-mile EPZs and to assess possible variations of emergency response for the two sites studied. These variations include evacuation and sheltering of population groups outside the 10-mile EPZ to a distance of 20 miles from an NPP. It is not expected that areas beyond the EPZ would need to take protective actions, but if they did, the protective actions would be limited to areas based on plume projections.

To support the treatment of evacuation of an EPZ, the project used site-specific evacuation time estimates (ETEs). For assessment of movement of the public residing between 10 to 20 miles, outside the EPZ, additional ETEs were developed for each site. The level of detail in developing these ETEs was significant, including the identification of general public and special facility population groups, e.g., hospitals, nursing homes, and prisons.

did we ultimately use these?

also subsequent pages

If not, take out

For nuclear power plants, Appendix E of 10 CFR 50 Section IV requires that an analysis of the time required to evacuate and time for taking other protective actions be provided for various sectors and distances within the EPZ for transient and permanent residents. An ETE is developed by licensees to support this requirement and is a tool that provides 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.

Protective actions beyond the EPZ ~~are~~ required by regulation but are not normally practiced. It is assumed that OROs would identify the need for such protective actions and direct that they be implemented in an ad hoc manner. Implementation of protective actions beyond the EPZ would be based on dose-projection data available to response organizations. Emergency response organizations would be aware of source terms and resultant doses that could require protective actions beyond 10 miles. The timing of emergency response actions was developed in accordance with the ORO plans.

Advancements in consequence modeling, specifically the development of WinMACCS, now allows detailed integration of protective actions into consequence analysis. WinMACCS allows temporal and spatial elements of sheltering and evacuation to be modeled. To use the features in WinMACCS, an ETE for the 10 to 20 mile zone was developed using the Oak Ridge Evacuation Modeling System (OREMS). OREMS was used to develop ETEs for the general public within the 10 to 20 mile zone for all sites. The output from OREMS was then used to support input into the WinMACCS model.

For each accident sequence that resulted in a release to the environment such that protective actions would be implemented, a baseline case was modeled. The baseline case included evacuation of the public residing in the 10-mile EPZ, a 20 percent shadow evacuation of the public residing in the 10 to 20 mile zone outside the EPZ, and sheltering of the remaining public within the 10 to 20 mile zone outside the EPZ for a period of 24 hours followed by evacuation. Selected alternative analyses and sensitivity analyses were also conducted.

6 more next page

Population subgroups, called cohorts, were defined to provide greater fidelity in the treatment of emergency response. For each site, six cohort groups were established. The makeup of the cohort groups varied by site depending on the population distributions and emergency management actions. As a general assumption, the accident scenario was assumed to occur during school hours, thus one cohort was established for schoolchildren within the EPZ. Other cohorts included the general public within the EPZ, general public in the 10 to 20 mile zone, special facilities within the EPZ, shadow evacuees, and a non-evacuating cohort.

Part of the characterization of emergency response is based on the timing of actions by onsite and offsite response organizations to protect public health and safety, generally by instructing the public to evacuate or shelter. The initiating event for many of the accident scenarios considered by SOARCA is a large earthquake close to the plant site. ~~The potential~~ for an earthquake is estimated in general by the occurrence of an earthquake ~~in the~~ past. ~~Understanding~~ the ~~location~~ of faults in the ~~eastern~~ United States is far from precise. Compared with the ~~situation~~ in the West where geological ~~fault lines can~~ be identified on the surface, faults ~~in the~~ east ~~are usually~~ buried below layers of ~~soil~~ and rock ~~and are~~ difficult ~~to identify, making~~ prediction of earthquake location and ~~magnitude~~ imprecise. not to down BP

The earthquakes hypothesized in SOARCA (30-40 km from Psite) are close to a plant site, and it can be assumed that severe damage is generally localized. Since the fault lines do not intersect the surface, most roads are not expected to be damaged but may be blocked by debris. Long-span bridges close to a site are unlikely to survive the earthquake. Thus, they are assumed to be impassible during emergency response. Hous~~ing~~es ~~stock~~ would generally survive the earthquake, but with some damage. The local electrical grid is assumed to be out of service due to the failure of lines, switchyard equipment, or other failures. There is no back up power system for the sirens at Peach Bottom, so they would be unavailable. Offsite response organizations would have to perform route alerting to notify the population of the need to take protective actions. This is a routine and effective method of informing the public and implementing protective actions [45]. It consists of the police driving through neighborhoods using bullhorns or other amplification devices to notify residents of the emergency.

5.3.1 Baseline Analyses of Emergency Response C

F~~or each accident sequence that resulted in a radioactive release to the environment that would invoke protective actions, a baseline case was modeled~~. The baseline case represents the protective action planning in place for EPZs [44]. Initial protective actions, for which guidance is provided in Supplement 3 to NUREG – 0654/FEMA-REP-1, Rev.1 [44], would likely include evacuation of the 2-mile zone around the NPP and evacuation of a 5-mile downwind keyhole, as shown in Figure 16. ~~The baseline analyses includes evacuation of the public residing within the 10-mile EPZ, a 20-percent shadow evacuation of the public residing in the 10 to 20 mile zone outside the EPZ, and sheltering of the remaining public within the 10-to-20-mile zone outside the EPZ.~~ A shadow evacuation is a spontaneous evacuation that is not specifically directed by the OROs. The population beyond 20 miles is assumed not to evacuate. Instead, this segment of the population is relocated if projected doses exceed EPA guidelines, ~~as discussed previously~~.

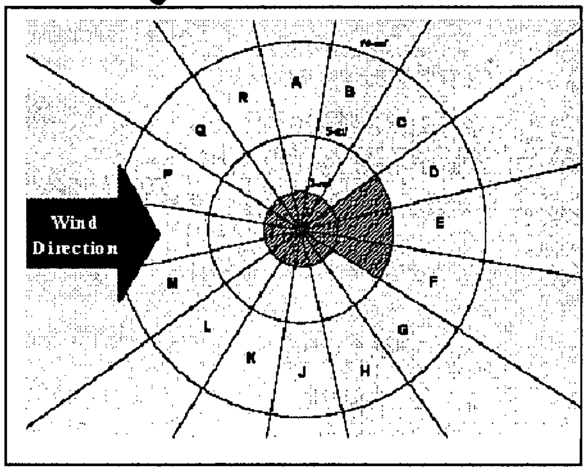

Figure 16 Standard Keyhole Evacuation

for consistency studies

The site specific ETEs were used to establish the evacuation parameters for the EPZ cohorts. To establish realistic evacuation parameters for the cohorts in the 10 to 20 mile zone, the evacuation was modeled using OREMS Version 2.6. OREMS is a Windows-based application used to simulate traffic flow and was designed specifically for emergency evacuation modeling [46]. Calculated adjustments to speeds of evacuation were made on an area segment basis to account for road condition differences (bottle-necks)

5.3.2 Sensitivity Analyses of Emergency Response

After completion of the baseline analysis, two variations were conducted, a 16-mile evacuation and a 20-mile evacuation for selected accident sequences. For the 16-mile analysis, complete evacuation of the 16-mile radius is assessed. The members of the public in the 16 to 20 mile zone were assumed to shelter for a period of 24 hours after containment failure, at which point this population group also evacuates.

Assessment of a complete evacuation within 20 miles from the plant was also conducted for selected accident sequences. For this assessment, an ETE was developed for the 20-mile area to provide realistic timing for the movement of the public in the treatment of consequences.

5.3.3 Integration with Consequence Modeling

WinMACCS was used to integrate the emergency-planning protective actions into the overall consequence modeling. WinnMACCS allows for the movement of different population groups, ~~referred to as cohorts,~~ and accommodates speed and direction variations of the evacuating cohorts. To fully utilize the functions of WinMACCS, the evacuation routes were assessed to determine the directions that evacuees would take. The evacuation area was mapped onto a grid with 64 compass sectors and 15 radii. This grid was used as the basis for the network evacuation model in WinMACCS. The same WinMACCS evacuation network was used for all accident sequences at each site. Only timing and evacuation speed parameters were adjusted to account for the specifics of each accident sequence.

discussed more fully in ~~preceding~~ section MELCOR 4.4.4

5.4 Source Term Evaluation from MELCOR to MACCS2

Source term evaluation for each of the accident sequences was performed using MELMACCS [47]. MELMACCS reads a MELCOR plot file and extracts information useful for source term definition for MACCS2. A number of user options have to be selected when using MELMACCS. 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 standard set of fission product groups, i.e., the Xe, Cs, Ba, I, Te, Ru, Mo, Ce, and La groups, are all included in the analyses. 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, ~~an~~ ORIGEN calculation was performed for SOARCA to estimate the inventory at mid-cycle, for which peak-rod burnup is estimated to be 49 MWd/kg. These data were used in MELMACCS to specify the inventory for MACCS2 and the MACCS2 input is, therefore, consistent with the MELCOR calculation. An analogous calculation was not performed for Surry; instead, a previously available fission product inventory based on the regulatory limit of burnup, 65 MWd/kg for the peak fuel rod, was used. This inventory should be ~~conservative in the sense of being~~ an overestimate~~s~~, at least for most of the fission products that do not reach secular equilibrium by mid-cycle.

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 MELCOR analyses used in SOARCA use reactor shutdown as the reference time, so the time of accident initiation is always set to zero in the MELMACCS input.

Aerosol deposition velocities are calculated by MELMACCS based on the geometric mean diameter of each aerosol bin, as defined in the MELCOR analysis. The deposition velocities are based on expert elicitation data using the median value of the combined distribution from the experts [48]. Typical values for surface roughness and mean wind speed, 0.1 m and 2.2 m/s, respectively, are additional parameters used to determine the deposition velocities in MELMACCS. Mean wind speeds were determined from the specific weather files used in the consequence analyses.

a very small fraction (<1%)

Finally, significant releases were broken up into one-hour plume segments. Trivial releases, such as those where the release fractions are less than ~~1 percent~~ of the core inventory and mostly noble gases, were sometimes broken up into longer time intervals.

5.5 Types of Site-Specific Parameters Used

Weather data for each site are taken from meteorological archives provided by each plant. The raw data were processed into 64 compass sectors in order to use the angular resolution capabilities in WinMACCS 3.4 and MACCS2 2.4.

Site files were processed with SECPOP2000 [49]. Population data were scaled by a factor of 1.0533 to account for US average population growth between the years 2000 and 2005. Economic data were scaled by 1.0900 to account for inflation between the years 2002 and 2005. The inflation adjustment was based on the increase in the consumer price index (CPI).

Site files were initially created by SECPOP2000 for 16 compass sectors, which is the only angular resolution supported by that code. WinMACCS was then used to interpolate these site files onto the 64 compass-sector grid that was used for the consequence analyses.

Consequence analyses were performed with WinMACCS using the standard approach of evaluating accidents in the following two phases:

1. Emergency phase is the period of time beginning with the initiating event and continues for about 1 week. The release from the plant and plume transport through the MACCS2 grid occur during this phase. Emergency response, i.e., evacuation and relocation of the population in order to reduce exposures and doses, also occurs during this phase.
2. Long-term 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.

Shielding factors applied to evacuation, normal activity, and sheltering for each relevant dose pathway (i.e., inhalation, deposition onto skin, cloudshine, and groundshine) were evaluated for each site based on values used in NUREG-1150. One departure from the NUREG-1150 values is for normal activity. Each of the normal activity values was reevaluated assuming that the average person spends 19 percent of the day outdoors and 81 percent of the day indoors [44]. The value for each of the pathways was evaluated as a linear combination of 19 percent of the value for evacuation and 81 percent of the value for sheltering.

for Surry, Virginia

Site-specific values are used to determine long-term habitability. Most states adhere to EPA guidelines that allow a dose of 2 rem in the first year and 500 mrem per year thereafter. The EPA recommendation has traditionally been implemented in MACCS2 as 4 rem during the first 5 years (2 rem + 4 * 0.5 rem) of exposure and that convention is adopted here. Some states, like Pennsylvania, have a stricter habitability criterion, 0.5 rem/yr beginning in the first year. Thus, the habitability or return criterion is somewhat site specific and is discussed further in Volumes ~~III~~ II and ~~IV~~ III.

~~EPA is in the process of adopting the habitability criterion defined by the Department of Homeland Security (DHS), which will allow a larger role for state and local governments to determine what is acceptable. Since census, economic, and public policy parameters used in SOARCA are based on the year 2005, more recent developments in EPA practices are not included in this study.~~

Other site-specific parameters include farmland and nonfarm-land values. These are also scaled from NUREG-1150 values using CPI as the basis for price escalation.

5.6 Reference to the Other, Non-Site Specific Parameters

There are a number of parameters used in the SOARCA analyses that are not site specific. They are described in the following paragraphs.

Ingestion of contaminated food and water is not treated in the SOARCA analyses. The reasoning is that adequate supplies of food and water are available in the US and can be distributed to areas affected by a reactor accident. Some farm areas would be taken out of production, at least for a period of time, while other areas would be put into production to compensate and maintain a level food supply without needing to resort to consumption of contaminated food. Likewise, bottled or filtered water from uncontaminated areas would be distributed to affected areas so that no one would need to consume contaminated water.

Some states have distributed potassium iodide (KI) tablets to people who live near commercial nuclear power plants. 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, the 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 (shortly before to immediately after plume arrival) so the efficacy is only 70%, ~~i.e.~~ that is, the thyroid dose from inhaled radioiodine is reduced by 70%. which would be

Much of the non-site-specific data used for consequence analysis in SOARCA are taken from a set of reports that document a joint NRC/Commission of the European Communities (CEC) expert elicitation study [48]. The data taken from this study include atmospheric dispersion parameters, dry deposition velocities, wet deposition parameters, and acute health-effect parameters. In all cases, the median values extracted from the elicitation study [48] are used for point-value consequence analyses in SOARCA. The processing of the NRC/CEC study as discussed in the section 5.3 is discussed in reference [ ] —

Evacuation was modeled ~~within a 10-mile emergency~~ planning zone (EPZ) at both sites. ~~Sensitivity analyses were also performed to determine the benefit of an ad hoc expansion of the EPZ to 16 and 20 miles for selected accident scenarios~~. Outside of the EPZ, the population was assumed to relocate if the projected dose exceeded a set of two upper bounds. These bounds were based on a range of dose levels published by the EPA, which is 1 to 5 rem. In SOARCA, the lower limit of this range, 1 rem, was used to trigger normal relocation and the upper limit of this range, 5 rem, was used to trigger hot-spot relocation. ~~In MACCS2, hot-spot relocation is performed first and normal relocation second~~. assumed to be The choices of times associated with normal and hot-spot relocation depended on the specific accident scenario because the first priority of
during the ~~early~~ phase of the accident emergency —

emergency responders is generally to evacuate those within the EPZ. ~~So~~ it was assumed that hot-spot relocation would ~~commence~~ begin sometime after evacuation was complete. and normal relocation would follow later.

The dose conversion factors (DCFs) used in the SOARCA analyses are based on Federal Guidance Report (FGR)-13 [50]. However, the standard DCFs were modified according to recommendations provided by Keith Eckerman [51]. One set of recommendations was to change 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 way the risk factors for cancers associated with those organs were evaluated. The two organs are bone marrow and breast~~s~~; 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. Keith Eckerman also recommended using dose to the pancreas as a surrogate for dose to soft tissue to estimate residual cancers. 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.

Keith Eckerman [51] 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 [52] and are consistent with the modified DCF file described in the preceding paragraph. These risk factors include 7 organ-specific cancers plus residual cancers that are not accounted for directly. reported in SOARCA

Decontamination parameters are based on values from NUREG-1150. Two levels of decontamination are considered, just as in NUREG-1150. The cost parameters associated with decontamination are adjusted to account for inflation using the CPI. Costs associated with a reactor accident are not ~~considered in this report~~; 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 Reporting Latent Cancer Fatality (LCF) Health Effects

Experts generally agree that it is difficult to characterize cancer risk for some organs because of the low statistical precision associated with relatively small numbers of excess cases. This limits the ability to estimate trends in risk. From an epidemiological standpoint, in most if not all cases, the number of latent cancer fatalities (LCFs) attributable to radiation exposure from accidental releases from a severe accident would not be 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 (WHO) estimated that 16,000 European cancer deaths will be attributable to radiation released from the 1986 Chernobyl nuclear power plant 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. Furthermore, WHO concluded that, "it is unlikely that the cancer burden from the largest radiological accident to date could be detected by monitoring national cancer statistics."

Four relevant studies have recently been completed:

New findings have been published from analyses of fractionated or chronic low-dose exposure to low, linear energy transfer (LET) radiation: ~~in particular~~; 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 ~~have recently been performed~~. Cancer risk estimates in these studies are generally compatible with those derived from the Japanese atomic bomb data. 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.
(See ... viii)

In the absence of additional information, the International Commission on Radiological Protection (ICRP), the National Academy of Science, and the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) have each indicated that the current scientific evidence is consistent with the hypothesis that there is a linear, no threshold (LNT) dose response relationship between exposure to ionizing radiation and the development of cancer in humans.

Conversely, the French National Academy of Medicine, in "Dose-effect relationships and estimation of the carcinogenic effects of low doses of ionizing radiation," March 30, 2005 [53], p. 1, advocates the following:

> 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) 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.

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

Ultimately, external and internal exposures to individual members of the public are converted from collective organ dose to LCFs using MACCS2. T~~he LNT model raises the concern that the summation of trivial exposures may inappropriately attribute LCFs to individuals far from the site of the accident~~. While the possibility of LCFs from very low doses cannot be ruled out, with low doses 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. Furthermore, in situations involving trivial exposures to large populations, ICRP and NCRP have noted that the most likely number of excess health effects is most likely zero, when the collective dose to such populations is equivalent to the reciprocal of the risk coefficient (about 20 person-Sv (2000 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), ICRP stated the following [54]:

> 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 regarding situations where collective dose should not be used, it did not provide guidance regarding 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 [52]:

> 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 go 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 offsite health consequences are attributable to very low radiation exposure. ~~In its most recent recommendations (ICRP Report 103), 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, but it did not explicitly provide a quantitative range for which exposures should not be considered.~~ ~~However,~~ In ICRP Report 104, "Scope of Radiological Protection Control Measures" [55], 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 millirem (mrem)) per year whole body dose. The International Atomic Energy Agency (IAEA) has stated that an individual dose is likely to be regarded as trivial if it is of the order of some several millirems per year. Although there is no scientific basis for defining a trivial dose, the ICRP and IAEA definitions of trivial dose may provide a basis to address truncation of offsite radiation exposure and the attribution of health consequences.

the Health Physics Society

Alternatively, (HPS) developed a position paper, "Radiation Risk in Perspective," revised August 2004 [56], 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 only be qualitative, discuss~~es~~ a range of possible outcomes, and emphasize~~s~~ the inability to detect any increased health detriment. The difference between the HPS view and those expressed by ICRP and IAEA is the detectability of a health consequence versus the difficulty of assessing the effects of exposure to trivial doses.

~~As discussed above,~~ The LNT model provides a viewpoint that is consistent with the regulatory approach of the NRC ~~which is based on LNT~~ and with past analyses using the MACCS2 code ~~have assumed an LNT dose response model~~. Additionally, these past analyses (e.g., NUREG-1150) calculated LCFs to 1,000 miles with forced deposition to account for all non-inert radionuclides in the dose calculation. Continued use of the LNT model provides consistency and comparability with previous work. The NRC is neither changing nor contemplating changing radiation protection standards and policy as a result of an approach taken in this study to characterize offsite health consequences for low probability events. On the other hand, the NRC can use different approaches for different applications. Therefore, the SOARCA analyses consider a range of dose truncation values, ranging from LNT on one hand to the Health Physics Society recommendation (5 rem/yr and 10 rem lifetime) on the other hand. Two intermediate dose-truncation levels are also considered. One is the 10 mrem/yr dose truncation value suggested ICRP Report 104; the other is US-average background radiation of 620 mrem/yr.

and is in accordance with the conclusions of national and international bodies

Results for these four dose-truncation levels are reported ~~without bias~~ for each of the accident scenarios considered in the SOARCA study, and, we believe, fairly represents the uncertainty in this area.

The statistic that is chosen to convey the likelihood of LCFs resulting from an accident at a NPP is the mean, population-weighted, individual risk. This value is more meaningful than the predicted number of LCFs ~~in the sense that it is representative of similar NPPs operating in areas with different population densities~~. The term "population-weighted" carries the meaning of the effect of population distribution, along with wind rose probabilities, on 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 arithmetic averages over the annual weather data used in the analysis. The initial phase of the SOARCA analyses only considers uncertainty in the weather; subsequent uncertainty analyses will consider the effect of source term and other input uncertainties on the predicted consequences. ~~In the subsequent uncertainty analyses, "mean" will represent the arithmetic average in a broader sense~~.

Mean, population-weighted, individual risks are presented at three ~~primary~~ distance ranges chosen to provide ~~that have historical significance~~ a pathway. The first range is 0 to 10 mi; the second is 0 to 50 mi; the third is 0 to 100 mi. The first distance range represents the population within the EPZ. The range from 0 to 50 mi is generally used in severe accident mitigation alternative (SAMA) and severe accident mitigation design alternative (SAMDA) analyses, and approximate the population in the food pathway EPZ. The range from 0 to 100 mi was chosen to demonstrate consequences out to a relatively long distance.

in that it ~~allows~~ facilitates comparison with expected cancer fatalities from all other causes in the same populations

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