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PKG ML090340075
ML090340081
ML090400077
ML090340092
ML090720110

POLICY ISSUE
(Information)

March 25, 2009 SECY-09-0045

FOR: The Commissioners

FROM: Brian W. Sheron, Director
Office of Nuclear Regulatory Research

SUBJECT: STATE-OF-THE-ART REACTOR CONSEQUENCE ANALYSES – PEACH BOTTOM AND SURRY RESULTS

PURPOSE:

The purpose of this paper is to inform the Commission of the results of the State-of-the-Art Reactor Consequence Analyses (SOARCA) for the Surry and Peach Bottom plants and to provide the Commission with the revised communication plan and a risk communication information booklet summarizing the SOARCA program for internal and external stakeholders. This paper does not identify any new commitments or resource implications.

SUMMARY:

The staff has completed an assessment of the Surry and Peach Bottom plants. The assessment of mitigation measures, including emergency operating procedures, severe accident management guidelines, and security-related mitigation measures, led the staff to conclude that all of the identified severe accident scenarios could reasonably be mitigated.

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Scenarios were also analyzed assuming all mitigation was not successful. The related accident progression and offsite consequence analyses confirmed that accident progression proceeds more slowly, offsite radiological releases are smaller and offsite consequences are less severe than indicated by earlier conservative and simplified analyses (e.g., NUREG/CR-2239, "Technical Criteria for Siting Criteria Development," commonly referred to as the 1982 Sandia siting study). The staff plans to complete the documentation of the current analyses in May 2009 and to initiate an external peer review and an uncertainty analysis. Completion of all activities and public release of information is planned for February 2010.

BACKGROUND:

In SECY-05-0233, "Plan for Developing State-of-the-Art Reactor Consequence Analyses," dated December 22, 2005, the staff proposed a plan to perform an updated realistic evaluation of severe reactor accidents and their offsite consequences. The staff indicated its intent that these analyses would reflect the accumulated improved understanding of severe accident behavior and potential consequences developed through the considerable research conducted by the U.S. Nuclear Regulatory Commission (NRC) and others over the last 25 years, and that the analyses would provide a body of knowledge on the more likely outcomes of such remote events. This information would be the basis for communicating that aspect of nuclear safety to Federal, State, and local authorities, licensees, and the general public. The staff also indicated that SOARCA would update quantification of offsite consequences documented in earlier studies (e.g., NUREG/CR-2239) that in some cases was based on overly conservative assumptions and simple bounding analyses to the extent that the earlier results are also overly conservative and can be misleading.

In a Staff Requirements Memorandum (SRM) dated April 14, 2006, the Commission approved the staff's plan and provided additional guidance in a number of areas. The Commission specifically concurred with the staff's approach to (1) use state-of-the-art analytical tools for accident progression and consequence analyses; (2) credit the use of Severe Accident Management Guidelines (SAMGs) and other new plant procedures, such as mitigative measures resulting from B.5.b (EA-02-026, Commission Order "Interim Safeguards and Security Compensatory Measures", February 25, 2002), and other like programs; and (3) use realistic site-specific evacuation scenarios and emergency planning modeling along with updated population and meteorological data. A summary of the staff's approach and results is provided in Enclosure 1. In this SRM, the Commission also directed the staff to develop communication techniques that could improve our communication of these complex analyses to the public. The communication techniques should address the role of mitigative strategies, identify important analysis assumptions, and discuss differences between the state-of-the-art analyses and the earlier analyses in the 1982 Sandia siting study. In response to this SRM item (and the subsequent SRM-COMSECY-06-0064 and SRM-SECY-08-0029 on this matter), the staff has undertaken a substantial effort to improve our risk communication of these severe low probability events as an integral part of the SOARCA project. The resulting revised communication plan (Enclosure 2) and risk communication information booklet (Enclosure 3) that utilizes current risk communication techniques for reporting the SOARCA results are provided.

In the April 2006 SRM, the Commission approved the staff's plan to focus on scenarios with a radiological release frequency greater than 10^{-6} per reactor year. The Commission also directed the staff to consider the potentially risk significant but lower frequency scenarios (e.g., the interfacing systems loss-of-coolant accident [LOCA] scenarios that bypass the containment). In response to this item, the staff modified its criterion for selecting scenarios to include events that bypass the containment with a core damage frequency (CDF) greater than 10^{-7} per reactor year. The staff also elected to use CDF as the metric for assessing nonbypass scenarios rather than radiological release frequency. This was a practical consideration (CDF values are available from Level 1 Probabilistic Risk Assessment (PRA) models). The staff briefed the Advisory Committee on Reactor Safeguards on the scenario selection process and adjusted the process to resolve their comments. While the objective of SOARCA was not to perform a level 3 Probabilistic Risk Assessment (PRA), we have confirmed our conclusion that we are addressing the most relevant accident scenarios by performing additional calculations and comparing our scenario selection criteria to the most probable and risk-significant scenarios identified in NUREG-1150. The staff also compares the selected SOARCA scenarios against security related aircraft events, the results of which are discussed further in Enclosure 4.

In SRM-COMSECY-06-0064, dated April 2, 2007 the staff was directed to reduce the scope of the SOARCA project to not more than eight plants representing a spectrum of plant designs and was also directed to focus on a subset of the eight plants (e.g., a boiling water reactor [BWR] and a pressurized water reactor [PWR] plant) to resolve methodological and technical issues. The staff selected the Peach Bottom plant as the BWR representative and the Surry plant as the PWR representative for the first assessments, and these plants are the focus of this paper and enclosures. The plant staffs at both facilities were very cooperative and provided plant-specific information and facility tours, without which we could not have completed this study. The staff has the cooperation of one additional plant, the Sequoyah plant, but has suspended the analysis of the Sequoyah plant pending the results and insights expected from the external peer-review of the Peach Bottom and Surry results. In this SRM, the Commission also reiterated its direction to the staff to use improved risk communication techniques. As stated previously, the SOARCA project has devoted significant efforts in that regard and has actively engaged the Office of Public Affairs (OPA) in developing the revised (and enclosed) communication plan and in review of the SOARCA risk communication information booklet.

In an April 3, 2007, memorandum to the Commission, "Treatment of Land Contamination and Offsite Economic Consequences in the SOARCA Project," the staff informed the Commission that significant technical limitations exist to the current economic models for calculating land contamination consequences. The Commission directed the staff in SRM-COMPBL-08-0002/ COMGBJ-08-003, "Economic Consequence Model", dated September 10, 2008, to address the economic consequence modeling outside of the SOARCA project in a separate initiative. The staff is developing an options paper for Commission consideration. Therefore, SOARCA consequence calculations are in terms of human health effects, including prompt and latent cancer mortality risk for specific events. The staff is pursuing the issue of economic consequences separately.

In SECY-08-0029, "State-of-the-Art Reactor Consequence Analysis – Reporting Offsite Health Consequences," the staff outlined a number of options for reporting predicted latent health effects and recommended an approach to assess and report latent health effects as the probability of an average individual's death from cancer (related to accident-related radiological releases) conditional to the occurrence of a severe reactor accident. The calculation would include health effects modeling assuming the linear no threshold (LNT) and 100 μSv (10 mrem) truncation dose response models, with results presented for three distances: (1) 0 to 16.1 km (10 miles), (2) 0 to 80.5 km (50 miles), and (3) 0 to 161 km (100 miles). The primary intent of this recommendation was to improve risk communication by communicating results in a way that could be compared to the occurrence of cancer fatalities in the general population from causes other than a reactor accident. In an SRM dated September 10, 2008, the Commission approved the staff's recommendation for assessing and reporting latent health effects and directed the staff to continue to coordinate with NRC's Federal partners as consequence modeling technology evolves. The staff has proceeded to assess and report the results accordingly. In addition, the staff has included supplementary sensitivity analyses using two additional dose response models—a dose response model that truncates health effects below 360 mrem per year (akin to normal background dose rate) and a model based on the Health Physics Society position paper, "Radiation Risk in Perspective," which does not quantify health effects below 5 rem in a year or 10 rem in a lifetime. We have performed these additional analyses in an effort to provide more perspective on potential outcomes and to assist in risk communication. In the SRM dated September 10, 2008, the Commission also approved the staff's recommendation to submit the Peach Bottom and Surry methodology and approaches for peer review by an external group of experts.

DISCUSSION:

In preparation for the detailed, realistic modeling of accident progression and offsite consequences, the staff had extensive cooperation from the licensees to (1) develop high-fidelity plant systems models; (2) define operator actions, including the most recently developed mitigative actions; and (3) develop models for simulation of site-specific and scenario-specific emergency planning. In addition, the staff met with the licensees and performed tabletop exercises with senior reactor operators, PRA analysts, and other licensee staff to gather information concerning scenario frequencies in their own PRAs and to establish the timing and nature of operator actions to mitigate the selected scenarios.

The staff identified two major groups of accident scenarios when we applied our scenario selection process using updated and benchmarked Standardized Plant Analysis Risk (SPAR) models and the best available plant-specific external event information. The first group, common to both Peach Bottom and Surry, was comprised of events commonly referred to as station blackout (SBO) scenarios, which include variations identified as short-term and long-term SBOs. 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 (dc) 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. Because the initiators were not always well differentiated in external events PRAs, the SBO was assumed—for the purpose of SOARCA analyses—to have been initiated by a seismic event, which is conservative because the seismic initiator was judged to be the most severe initiator in terms of timing, with respect to the system failures occurring at the beginning of the scenario. 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 similarly 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 scenario involving an interfacing systems LOCA (ISLOCA) due to an unisolated rupture of low-pressure safety injection piping outside containment and the second scenario involving a thermally induced steam generator tube rupture. The latter occurs as a variant of an SBO scenario. Again, these scenarios are generally understood to be important potential contributors to risk in PRAs.

The assessment of mitigation measures, including emergency operating procedures, severe accident management guidelines, and security-related mitigation measures, led us to the conclusion that all the identified scenarios could reasonably be mitigated. The security-related measures to provide alternative ac power and portable diesel-driven pumps were especially helpful in counteracting SBO scenarios. For the ISLOCA sequence, 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 the one event involving core damage, the Surry short-term SBO, the mitigation was sufficient to enable flooding of the containment through containment spray systems to cover core debris. The assessment of the mitigation measures was undertaken with support from integrated accident progression analyses using the MELCOR code, which incorporates our best understanding of plant response and severe accident phenomenology. MELCOR analyses were used to both confirm the timing 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 were performed crediting only installed equipment. These analyses resulted in no core damage and revealed that success criteria in many PRAs are overly conservative. The SOARCA study has revealed a number of insights that many existing PRAs should consider adopting to more realistically estimate the risk of nuclear power plant operations. These insights include: 1) credit for the prevention and mitigation of severe accidents using SAMGs and security-related mitigation measures, 2) more realistic success criteria for core cooling, 3) credit for delayed timing of significant core damage, and 4) credit for delayed timing and reduced magnitude of offsite releases. In parallel with the SOARCA study, the staff has been incorporating security-related mitigation measures into its Standardized Plant Analysis of Risk (SPAR) models, reassessing SPAR model success criteria, and expanding several SPAR models to address severe accident progression (i.e., the development of a limited scope Level 2 PRA capability). Currently, there is no plan to expand the SPAR models to provide offsite consequence estimates. Following completion of the SOARCA peer review, the staff will assess how to appropriately incorporate SOARCA study insights in the SPAR models.

To quantify the benefits of the mitigative measures 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 was unmitigated, leading ultimately to an offsite release. Overall, the MELCOR accident progression analyses confirmed that accident progression in severe accidents proceeds much more slowly than earlier conservative and simplified treatments indicated. The reasons for this are principally twofold. Research and development of better phenomenological modeling has produced results that show a much more protracted and delayed core degradation transient with substantial delays of reactor vessel failure. Furthermore consistent treatment of all aspects of accident scenarios including more complete modeling of plant systems also often yields delays in core damage and radiological release. In contrast, in past simplified treatments using qualitative logical models, bounding approaches have often been used that produce more conservative results.

In SOARCA, where initial conditions and analytical assumptions for the specific sequence are propagated throughout subsequent analysis and are analyzed in an integral fashion using MELCOR, it can be seen 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 long-term SBO, the most likely accidents considered in SOARCA (assuming no mitigation), 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 hrs (PWR) was available before the onset of offsite radiological release due to containment failure. In the 1982 siting study (in which the dominant event was identified as the Siting Source Term 1 release) it was assumed that a major release occurs in 1 ½ hours. The SOARCA analyses clearly indicate that ample time is available for operators to take corrective action even if initial efforts are assumed unsuccessful. Moreover, these time delays also allow substantial time for input from plant technical support centers and emergency planning. 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 hrs), reactor vessel failure is delayed for roughly 8 hours, allowing time for restoration of cooling and prevention of vessel failure. In these cases, containment failure and radiological release are 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, analyses show the radiological release to be substantially reduced.

The SOARCA study also demonstrated that the magnitude of the fission product release is likely to be much smaller than assumed in past studies. Again, this is a result of extensive research and improved modeling as well as integrated and more complete plant simulation. The study predicted typical releases of important radionuclides such as iodine and cesium to be no more than 10 percent and more generally in the range of 0.5 to 2 percent. By contrast, the 1982 siting study assumed an iodine release of 45 percent and a cesium release of 67 percent.

As the result of the accident progression and source term analysis, combined with realistic simulation of emergency planning, offsite health consequences are dramatically smaller than reported in earlier studies. Because of the delayed nature of the releases and their diminished magnitude, no early acute health effects were predicted, close-in populations were evacuated, and no early fatalities occurred. Latent health effects are also quite limited, even using the most conservative dose response treatment. In fact, much of the latent cancer risk for the close-in population was derived from the relatively small doses received by populations returning to their homes in accordance with emergency planning guidelines. 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, the risk to an individual for an important severe accident scenario is on the order of 10^{-9} to 10^{-10} per reactor year. In comparison, these risks are thousands of times smaller than the NRC safety goal and a million times smaller than the U.S. average risk of a cancer fatality.

The enclosures to this paper are the Executive Summary of the SOARCA NUREG, a revised communication plan for reporting results, an information booklet that utilizes the latest risk communication techniques for presentation of the results of SOARCA to internal and external stakeholders, and a separate safeguards enclosure that will be provided separately. The SOARCA communication activities have been coordinated with the Office of Public Affairs and communication staff from the Office of the Executive Director for Operations.

The staff plans to complete the NUREG documentation in May 2009 and the external peer review in January 2010. As a parallel effort, an uncertainty study will begin shortly to quantify the effect of epistemic and aleatory uncertainties on consequence estimates. Upon completion of this work, the staff will present these findings to the Commission along with options for their resolution and the staff's proposal to implement the communication plan.

COORDINATION:

The SOARCA project has been conducted as a coordinated effort involving the Office of Nuclear Regulatory Research, Office of Nuclear Security and Incident Response, Office of New Reactors, Office of Nuclear Reactor Regulation, and Office of Public Affairs. Moreover, the project was guided by a steering committee composed of senior managers from the above program offices. Regional offices have received interim briefings. The Office of the General Counsel reviewed this package and has no legal objection. The Office of the Chief Financial Officer reviewed this package and determined that it has no financial impact.

We request this SECY paper not be made publicly available because it contains pre-decisional, sensitive internal information pending the completion of the external peer-review and uncertainty analysis.

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Enclosures:
1. Summary of Results for Peach Bottom and Surry plants
2. SOARCA Communication Plan, Rev. 3
3. SOARCA Information Booklet
4. Security-Related Scenarios (provided separately)

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Enclosure 1

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) could be extremely conservative, to the point that predictions were not useful for characterizing results or guiding public policy. In some cases, the overly conservative results were driven by the combination of conservative assumptions or boundary conditions. In other cases, simple bounding analysis was used in the belief that if the result was adequate to meet an overall risk goal, bounding estimates of consequences could be tolerated. The subsequent misuse and misinterpretation of such bounding 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, it is the objective of this study that this updated plant analysis include the significant plant improvements and updates (e.g., system improvements, training and emergency procedures and offsite emergency response), which have been made by plant owners and are not reflected in earlier NRC assessments. These improvements to plant safety also include those enhancements recently made in connection with security-related events. Thus, a key objective 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 this evaluation 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, and NUREG/CR-2723, "Estimates of the Financial Consequences of Reactor Accidents," dated September 1982.

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. This analysis of Peach Bottom and Surry is being reviewed by the Advisory Committee on Reactor Safeguards and will be the subject of an external peer review.

Method

The approach was to utilize the detailed, integrated, phenomenological modeling of accident progression (reactor and containment thermal-hydraulic and fission product response) that is embodied in the MELCOR code coupled with modeling of offsite consequences (MACCS code) in a consistent manner (e.g., accident timing) to estimate offsite consequences for important reactor accidents. The approach is 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, together with the improved MACCS code and the incorporation of site-specific and updated sequence-specific emergency planning. 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, it was apparent that the demonstration of 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.

What sequences should then be analyzed to demonstrate the benefits of our improved understanding incorporated into detailed, best-estimate modeling and the many plant improvements that have been realized over the last 25 years? To efficiently achieve these objectives, it seemed logical that we should select sequences that result in substantial offsite releases but also reflect probabilistic considerations - focusing on the more credible yet low-frequency accident sequences. By this approach, we could avoid the needless quantification of many sequences that are extremely low in probability or pose only residual risk. Further, SOARCA is intended to provide perspective on the question, "What are the likely outcomes and what is our best estimate of the risk if a severe accident were initiated at a nuclear plant?" The updated SOARCA requantification of consequences might include consideration of those sequences important to risk as demonstrated by a full-scope level 3 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. Fortunately, the preponderance of level 1 PRA information, combined with our insights on severe accident behavior obviates the need for such information in selecting sequences. Ample PRA information 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 fission product release transport and deposition, allow us to utilize core damage frequency (CDF) as a surrogate screening 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 sequences will be captured.

The sequence selection criteria identify risk-significant sequences in both an absolute and relative sense. It can be shown (see Appendix D of NUREG-1860) that a core damage frequency (CDF) of 10^{-4} per reactor-year and a large early release frequency (LERF) of 10^{-5} per reactor-year are acceptable surrogates to the latent and early quantitative health objectives (QHO) contained in the Commission's Safety Goal Policy Statement [51 FR 28044]. 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 RG 1.200, defines a significant sequence as one of the set of sequences that, when rank ordered by decreasing frequency, aggregate to 95 percent of the CDF or that individually contribute more than 1 percent of the CDF. Therefore, the SOARCA sequence selection criteria are consistent with previously issued regulatory guidance. More importantly, they help to place severe accidents into their proper risk context. A search for high-consequence severe accidents, without consideration of the likelihood of their occurrence, can be an exercise that loses the perspective that one is entering a realm of very low residual risk, a realm where the risk quantification is suspect (often conservative) and may be more misleading than revealing.

Another yardstick for assessing the impact of low-frequency events is to consider the increase in the consequences that would be necessary to offset the lower frequency. Conceptually, an event with a larger radiological release could have greater risk if the increase in the radiation release is larger than the decrease in frequency of the event. For example, all other considerations equal, 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 risk. Since we are including events with substantial volatile releases on the order of 1 to 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 risk 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 along with 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 screening 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 conclusively 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 smaller or equivalent absolute risk than 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 fission product 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 general societal impacts of an extremely large earthquake—larger than that generally considered in residential or commercial construction codes (past or present)—such that a potentially significant impact would be had on the public health irrespective of the nuclear plant response. Such an analysis has not been performed for the areas surrounding the plants selected for SOARCA and would have to accompany an evaluation of nuclear plant risk to provide the perspective on the incremental 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 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 post-9/11 mitigation measures. Post-9/11 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 post-9/11 mitigation measures in December, 2008.

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 Fission Product 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 mitigation measures, including emergency operating procedures, severe accident management guidelines, and security-related mitigation measures, led us to the conclusion that all the identified scenarios could reasonably be mitigated. 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 timing 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 quantify the benefits of the mitigative measures and to provide a basis for comparison to past analyses of unmitigated severe accident scenarios, the SOARCA project analyzed these same groups of scenarios conservatively assuming the event proceeded as unmitigated and led ultimately to an offsite release.

<u>Offsite Radiological Consequences</u>

An independent expert panel was assembled to review the proposed severe accident modeling approach of MACCS to identify areas that would benefit from improvement. 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. Improvements made to the code and input files include use of 64 radial directions for plume travel instead of 16 as well as use of short (1 hour long) plume segments.

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 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 those individuals who may 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 about 20 percent of the public in the area from 16 to 32 km (10 to 20 miles) from the plant was assumed to evacuate as a shadow evacuation 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 fission product 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.

Metrics for the 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 assuming the linear-no-threshold model (LNT) 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. Dose truncation values were used to help provide insight into the latent cancer health effects relative to the dose received by different exposure groups. Inclusion of dose truncation values is not meant to imply any NRC endorsement of a truncation value. Rather, it allows various audiences to readily understand the calculated consequences in a context that resonates with their assumptions of the dose-response relationship. Dose truncation values used for SOARCA included 10 mrem/year representing a small dose, 360 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 commonly referred to as 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, which is conservative. 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 as well as reactor safety systems are similarly 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 LOCA (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	$2x10^{-6}$	Immediate loss of ac power and turbine-driven systems
Thermally induced steam generator tube rupture	Seismic, fire, flooding	$5x10^{-7}$	Immediate loss of ac power and turbine-driven systems, consequential tube rupture
Interfacing systems LOCA[1]	Random failure of check valves	$3x10^{-8}$	Check valves in high-pressure piping fail open causing low-pressure piping outside containment to rupture, followed by operator error

Mitigation Measures

The assessment of mitigation measures, including emergency operating procedures, severe accident management guidelines, and security-related mitigation measures, led us to the conclusion that all the identified scenarios could reasonably be mitigated. 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

[1] The licensee's PRA core damage frequency was $7x10^{-7}$.

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 timing 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.

Accident Progression and Fission Product Release

To quantify the benefits of the mitigation measures 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, leading ultimately to an offsite release. The overall result of the MELCOR accident progression analyses was the confirmation that accident progression in severe accidents proceeds much more slowly than earlier conservative and simplified 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 consistent treatment, which includes more complete modeling of plant systems and also often yields delays in core damage and radiological release. Bounding approaches have often 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, it can be seen 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 1982 siting study, for the dominant event (identified as the SST1 release), it was assumed that a major release occurs in 1½ hours. The SOARCA analyses clearly indicate that ample time is available for operators to take corrective action even if initial efforts are assumed unsuccessful. Further, these time delays also allow substantial time for input from plant technical support centers and emergency planning. 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 induced steam generator tube rupture	3	7.5	3.5
Interfacing systems LOCA	9	15	10

The SOARCA study also demonstrated that the magnitude of the fission product release is likely to be much smaller than assumed in past studies, again as a result of extensive research and improved modeling and as a result of integrated and more complete plant simulation. Typical releases of important radionuclides such as iodine and cesium are predicted to be no more than 10 percent and are more generally in the range of 0.5 to 2 percent. By contrast, the 1982 siting study assumed an iodine release of 45 percent and a cesium release of 67 percent. Figures 1 and 2 provide the fission product release results for iodine and cesium.

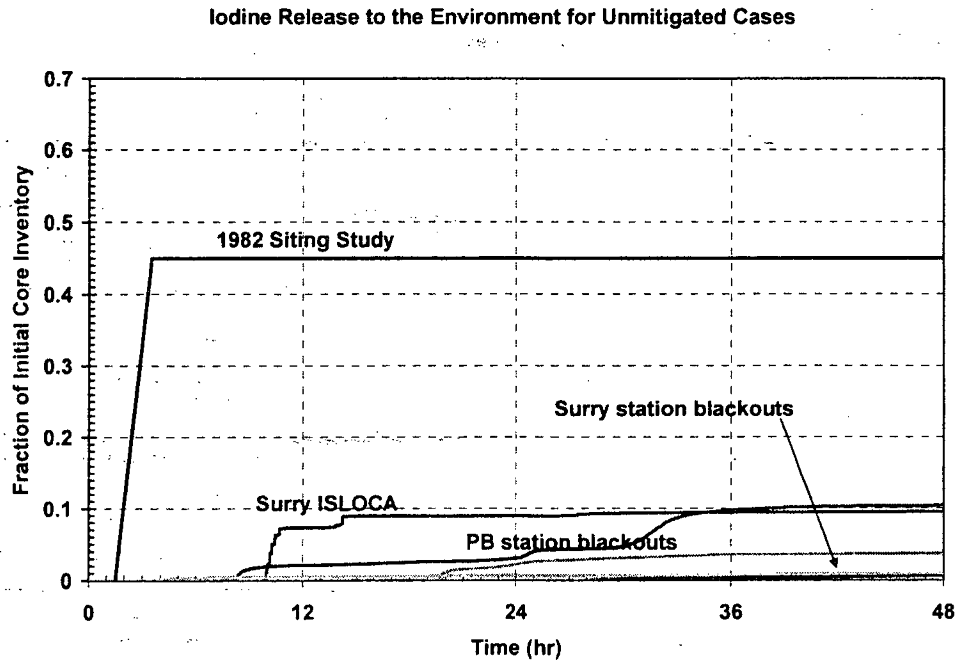

Figure 1. Iodine Releases to the Environment for SOARCA Unmitigated Scenarios

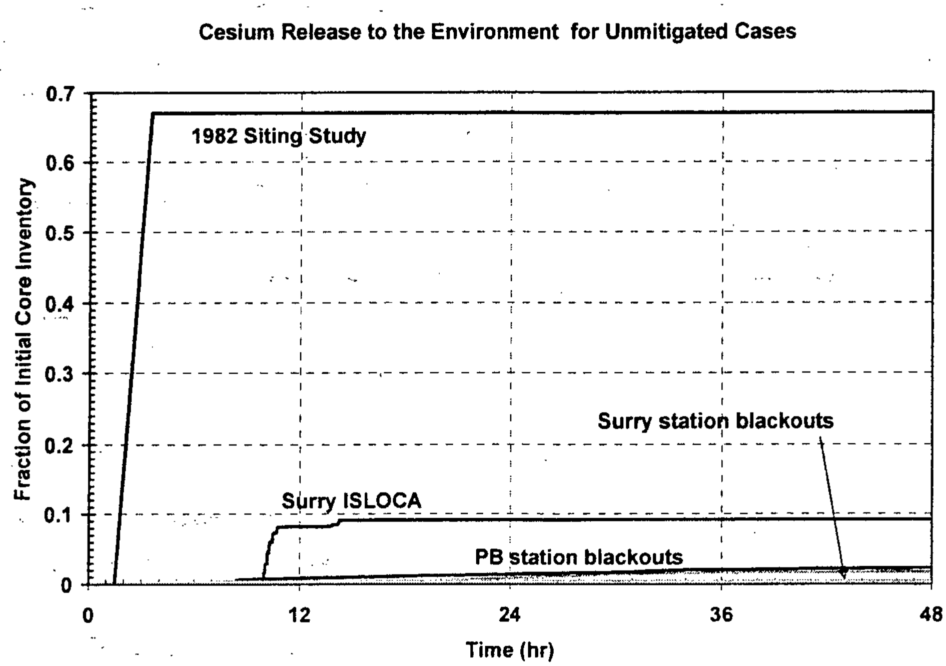

Figure 2. Cesium Releases to the Environment for SOARCA Unmitigated Scenarios

Sequences involving large early releases have influenced the results of past PRAs and consequence studies. For example, the 1982 Siting Study results were controlled by an internally initiated event with a large early release that was assigned a frequency of $1x10^{-5}$/year. 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 neglecting post-9/11 mitigation measures. 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.

Offsite Radiological Consequences

The result of the accident progression and source-term analysis, combined with realistic simulation of emergency planning, is that offsite health consequences are dramatically smaller than reported in earlier studies. Because of the delayed nature of the releases and their diminished magnitude, no early acute health effects were predicted; close-in populations were evacuated and no early fatalities occurred. Latent health effects are also quite limited, even using the most conservative dose response treatment. 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 linear-no-threshold (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, 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 thousands of times smaller than the NRC safety goal for latent cancer fatality risk of $2x10^{-6}$ per reactor-year and a million times smaller than the U.S. average risk of a cancer fatality of $2x10^{-3}$ per year. Tables 5 and 6 provide the risk estimates for individual SOARCA scenarios 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	$3x10^{-6}$	$2x10^{-4}$	$6x10^{-10}$
Short-term SBO	$3x10^{-7}$	$2x10^{-4}$	$7x10^{-11}$

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	$2x10^{-5}$	$5x10^{-5}$	$7x10^{-10}$
Short-term SBO	$2x10^{-6}$	$9x10^{-5}$	$1x10^{-10}$
Thermally induced steam generator tube rupture	$5x10^{-7}$	$3x10^{-4}$	$1x10^{-10}$
Interfacing systems LOCA	$3x10^{-8}$	$7x10^{-4}$	$2x10^{-11}$

To provide additional information on the potential range of health consequences, the SOARCA project has developed latent cancer risk estimates assuming the LNT 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 a small dose, 360 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. Tables 7 and 8 show the results of sensitivity calculations for dose truncation values for background and the Health Physics Society Position. 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 from the LNT model.

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 due to plume dispersion and fission product deposition closer to the site.

As noted above, the SOARCA offsite consequence estimates are dramatically 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 0 for both sites. Also, the Siting Study predicted 2,700 cancer fatalities for Peach Bottom and 1,200 for Surry for the SST1 source term using the LNT model. Although the exact basis for these cancer fatality estimates could not be recovered, literature searches and sensitivity analyses with MACCS suggested that these estimates are for the population within 500 miles of the site. Although SOARCA does not include the same latent cancer fatality consequence metrics as the Siting Study, an indirect comparison is possible. SOARCA predicted that the conditional risk of latent cancer fatality for an individual located within 10 miles assuming LNT was $2x10^{-4}$ for Peach Bottom and from $5x10^{-5}$ to $7x10^{-4}$ for Surry. Multiplying this conditional risk by the population within 10 miles of each site roughly corresponds to about 10 cancer fatalities for Peach Bottom and 10 to 100 for Surry for the population within 10 miles. SOARCA estimates for large distances would make the SOARCA predictions larger due to a larger exposed population in combination with the LNT assumption, while application of dose truncation criteria would make the SOARCA predictions smaller.

Table 7. Peach Bottom 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	$6x10^{-10}$	$3x10^{-11}$	$5x10^{-12}$
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	$7x10^{-10}$	$2x10^{-11}$	$2x10^{-14}$
Short-term SBO	$1x10^{-10}$	$1x10^{-11}$	$2x10^{-14}$
Thermally induced steam generator tube rupture	$1x10^{-10}$	$4x10^{-11}$	$3x10^{-12}$
Interfacing systems LOCA	$2x10^{-11}$	$8x10^{-12}$	$5x10^{-12}$

Enclosure 2

Communication Plan for the State-of-the-Art Reactor Consequences Analyses (DRAFT - Revision 3)

Overview

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, it is the objective of this study that this updated plant analysis include the significant plant improvements and updates (e.g., system improvements, training and emergency procedures and offsite emergency response), which have been made by plant owners and are not reflected in earlier NRC assessments. These improvements to plant safety also include those enhancements recently made in connection with security-related events.

The first phase of SOARCA analyzed examples of two major types of nuclear reactor in the United States: (1) Peach Bottom Atomic Station, a boiling water reactor (BWR), and (2) Surry Nuclear Power Plant, a pressurized water reactor (PWR). The first phase has been completed and a summary of the results was provided to the Commission. The staff is now developing a draft NUREG for peer review. Upon completion of the independent external peer-review, the staff will incorporate the peer-review comments and release the results of SOARCA in the form of a technical report (NUREG) and a risk communication information booklet (NUREG/BR). NRC will then consider whether analyses are needed for other reactor types and sites.

Goals

The goal of SOARCA is to determine best estimates of the offsite radiological consequences for severe accidents at U.S. operating reactors using a methodology based on state-of-the-art analytical tools and to present those results using risk communication techniques to achieve informed public understanding of the important factors. These factors include the extent and value of defense-in-depth features of plant design and operation as well as mitigation strategies that are employed to reduce risk. As a result, SOARCA will update analyses such as NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," dated November 1982.

Background

To develop information that will help in its regulatory mission to protect the public, NRC has performed several research studies to understand probabilities and potential consequences of severe accidents at nuclear plants. Because limited realistic information was available for these historical studies, they were based on conservative assumptions about how the plants would behave. These publicly available estimates have, at times, been misinterpreted and misused. To improve public understanding, the SOARCA project seeks to produce more realistic and likely estimates.

Over the past 25 years, NRC, industry, and international nuclear safety organizations have completed substantial research on plant response to hypothetical accidents that could damage the core and containment. That research has significantly improved NRC's ability to analyze and predict how nuclear plant systems and operators would respond to severe accidents. During that same time, reactor owners have improved plant designs, emergency procedures, maintenance programs, and operator training, all of which have enhanced plant safety. Plant owners and local governments also have refined and improved emergency preparedness measures to further protect the public in the event of a severe accident. The SOARCA team applied this accumulated research and incorporated plant enhancements to achieve a more realistic evaluation of consequences from severe nuclear accidents. The results of this research will become the foundation for communicating aspects of severe accidents and updating information from older research studies.

The NRC staff used state-of-the-art information and computer modeling tools to develop best estimates of accident progression and, for scenarios in which accidents proceed to core damage, what radioactive material could potentially be released into the environment. The staff then assessed those releases to realistically estimate the potential consequence to the public. The staff considered the following in these new analyses:

1. Design-specific reactor accident sequence progression, taking into account the plant's current design configuration.

2. Design-specific potential containment failure timing, location, and size.

3. Site-specific emergency planning assumptions, including evacuation and sheltering.

4. Credit for operator actions based on emergency operating procedures, severe accident management guidelines, and post-9/11 and other mitigation measures that were in place at the time of the assessment.

5. Site-specific meteorological conditions and updated population data.

The agency learned more about realistic accidents by rigorously and realistically quantifying a relatively few important events. The project set technical criteria to determine which scenarios were important and focused its resources accordingly. The project team included scenarios having an estimated core damage frequency of 10^{-6} per reactor year (one in a million) or greater. Also, bypass scenarios having an estimated core damage frequency of 10^{-7} per reactor year (one in 10 million) or greater were included.

As noted above, the accident analysis for each scenario included credit for operator mitigation actions. Also, to quantify the benefits of the mitigation measures and to provide a basis for comparison to past analyses of unmitigated severe accident scenarios, these same scenarios were analyzed in the SOARCA project assuming the event proceeded as unmitigated, leading ultimately to an offsite release.

An independent, external peer-review committee will examine the approach and underlying assumptions and results obtained for Peach Bottom and Surry to ensure that they are defensible and state-of-the-art.

Key Messages

General Messages

- In carrying out its mission to protect public health and safety, NRC performs research to determine the risk of commercial nuclear power plant operation to the public. The SOARCA research project realistically estimates the potential consequences to the public given the state-of-the-art understanding of accident phenomena and plant performance under accident conditions.

- The results of this project indicate reactor safety has improved over the years as a result of efforts by industry to improve plant design and operation and by NRC to develop improved regulations to enhance safety.

- The SOARCA cancer risk values are all significantly smaller than the NRC-established safety goal that "individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health."

- Both mitigated and unmitigated cases predict that no early fatalities will occur and average individual latent cancer fatality risks are very low for the unmitigated scenarios examined.

- Our analyses indicate that potential radiation releases would occur several hours later than earlier thought, and they would be substantially smaller; as a result, offsite consequences from severe accidents at nuclear power plants would be smaller than previously predicted.

- The results of this consequence analysis provide the public, NRC, and other government agencies with a more realistic picture and a better understanding of potential consequences in the unlikely event of an accident.

Additional Key Messages for the Scientific Community

- Information developed from years of research has been incorporated into the tools that NRC uses to evaluate potential accidents. These tools are the SPAR, MELCOR, and MACCS2 computer codes. These codes were used to select the scenarios, to model nuclear power plant systems and operator responses to severe accident conditions, and to produce a best estimate of consequences to the public.

- This study focuses on those accidents estimated to have a one in a million chance per year or greater of core damage (a core damage frequency of approximately equal to or greater than 10^{-6} per reactor year). SPAR models were used to identify those potential scenarios for further evaluation.

- In addition, the project placed emphasis on sequences that may be a little less likely to occur but with the potential for more severe consequences. Containment bypass events have the potential for more severe consequences and, therefore, those bypass sequences estimated to have a 1 in 10 million chance per year or greater to result in core damage (a core damage frequency equal to or greater than 10^{-7} per reactor year) were included within the scope of SOARCA. The project teams used the SPAR models to identify the included potential bypass scenarios.

- Plant-specific MELCOR analyses reflected design-specific features. MELCOR code modeled the nuclear power plant behavior, the progression of the accident, and the radioactive material released into the environment. This includes the timing of fuel damage, component failures, and releases to the environment.

- Structural analyses determined the expected containment performance during accidents.

- MACCS2 calculations used site-specific actions, emergency planning, weather data, population data, and evacuation times (including sheltering) to estimate consequences such as early fatalities and latent cancer injuries.

Communication Team

The communication team includes the following members and will be responsible for facilitating communication activities for the SOARCA project:

- Team Manager: Jimi Yerokun, Office of Nuclear Regulatory Research
- Team Members: Terry Brock, Office of Nuclear Regulatory Research
- Charles Tinkler, Office of Nuclear Regulatory Research
- Richard Guzman, Office of Nuclear Reactor Regulation
- Scott Burnell, Office of Public Affairs
- Susan Bagley, Office of the Executive Director for Operations
- David Decker, Office of Congressional Affairs

As the project progresses, it is expected that other NRC staff members will participate in communication activities, as needed.

Audiences

External Stakeholders include:

- General public
- Public interest groups
- Media
- Congress
- Licensees
- Nuclear industry organizations (e.g., Nuclear Energy Institute, Institute of Nuclear Power Operations, Electric Power Research Institute)
- Department of Homeland Security and other Federal and State agencies
- State regulators and Agreement States
- International groups

Internal Stakeholders include:

- The Commission
- Advisory Committee on Reactor Safeguards (ACRS)
- NRC staff

Communication Tools

The following tools will be used to communicate with external stakeholders:

Public Website	SOARCA information will be placed on the external website.
Questions and Answers	This contains information that highlights aspects of the project that audience members may inquire about. These questions and answers are given at the end of this Communication Plan.
Fact Sheet	A fact sheet will be prepared to provide the public with an overview of the project.
Information booklet	A summary of the SOARCA project will be presented in a separate NUREG/CR booklet using plain language and applying risk communication techniques. This booklet is a tool to enable NRC and its stakeholders to develop a common understanding about risk. It will be issued after the peer review is completed.
Public Meetings	Meetings will be held to publicly share information at key phases of the project.
Press Releases	A press release will be issued after the peer review is completed and at other times as appropriate. Press releases will be coordinated with the Office of Public Affairs.
Technical Reports	Technical information about the process and results will be documented in a NUREG and will be made publicly available through the Agencywide Documents Access and Management System (ADAMS) and the NRC's external website. This NUREG is being developed and will be issued after the peer review is completed.
External Briefings	Briefings will be provided to Congressional and State stakeholders as requested.
Internal Briefings	Prior to releasing the results, the SOARCA staff will hold briefings for technical staff in NRC Regional offices and other interested NRC staff to help prepare them to communicate about the results.

MILESTONES OF COMMUNICATION ACTIVITIES

COMMUNICATION ACTIVITIES	RESPONSIBILITY	DATE
Semi-annual TA brief	T. Brock	ongoing
Quarterly DEDO brief	T. Brock	ongoing
Meeting with ACRS	SOARCA Team	July 2009
Independent Peer Review of documents starts – issue press release	SOARCA Team/OPA	July 2009
Steering Committee meeting	T. Brock	September 2009
Public release of SOARCA results for Peach Bottom and Surry – The following activities are planned to engage stakeholders to promote a common understanding of the SOARCA results.		
Public website update	SOARCA Team	Early 2010
Briefings on results to Regional and HQ staff prior to public release (nonpublic meeting)	T. Brock	Early 2010
Briefings on results to participating licensees	T. Brock	Early 2010
All-Agreement States and Non-Agreement States letter	T. Brock/FSME/DILR	Early 2010
Press release to coincide with the release of the SOARCA results	T. Brock/OPA	Early 2010
Chairman holds press briefing	T. Brock/OPA	Early 2010
Public release of NUREG and the NUREG/BR information booklet	SOARCA Team/SNL/ OPA	Early 2010
Public Workshop	SOARCA Team/SNL	Early 2010
Regulatory Information Conference (RIC) Presentations on final results	SOARCA Team	March 2010

Evaluation and Monitoring

The communication plan continues to be updated to reflect key ideas being communicated to stakeholders and key decision points in the project's progress. Communication from these venues will be reflected in responses to key questions and ideas during the project's progress.

Questions and Answers

What is the State-of-the-Art Reactor Consequences Analyses (SOARCA) project?

SOARCA is a research project that develops realistic estimates of the potential public health effects from a nuclear power plant accident, where low-likelihood scenarios could release radioactive material into the environment and potentially cause offsite consequences. The project also evaluates and improves, as appropriate, methods and models for realistically evaluating both the plant response during such severe accidents, including evacuation and sheltering and the potential public risk.

Why is the U.S. Nuclear Regulatory Commission (NRC) performing this study?

NRC is doing this study to develop the most realistic evaluations possible for the potential consequences of severe nuclear accidents. Over the years, NRC, industry, and international nuclear safety organizations have completed substantial research on plant response to hypothetical accidents that could damage the core and containment. The results have significantly improved NRC's ability to analyze and predict how nuclear plant systems and operators would respond to severe accidents. Also, plant owners have improved the plant design, emergency procedures, maintenance programs, and operator training, all of which have improved plant safety. Emergency preparedness measures also have been refined and improved to further protect the public in the highly unlikely event of a severe accident. Combining all of this new information and analysis will improve the realism of accident consequence evaluations.

How will this study be different from earlier studies?

The SOARCA project will:

- Use an improved understanding of source terms and severe accident phenomenology.
- Credit the use of severe accident mitigation strategies and procedures.
- Use updated emergency preparedness modeling.
- Account for plant improvements.
- Use modern computer resources and advanced software to yield more accurate results.

In addition, the SOARCA project is designed to be a more realistic estimate. Some of the earlier studies also were designed to be best estimates; however, because they were limited by the available knowledge of accident phenomenology, these older studies were conservative (particularly the very improbable severe accidents). The SOARCA project will provide the latest basis from which the public and decision makers can assess the consequences of severe reactor accidents.

What are the potential uses of the SOARCA study?

The overarching purpose of this study is to provide more realistic information about potential nuclear power plant consequences to the public and other stakeholders, including Federal, State, and local authorities. This study also will increase understanding of the value of defense-in-depth features of plant design and operation, including the use of mitigative strategies.

What consequence measures are being estimated?

This study assesses the health effects of a potential radiation release to the general public. State-of-the-art analytical models estimate the individual risk of prompt fatality and latent cancer fatality that could occur in the remote event that a severe reactor accident occurs. Prompt fatalities are those resulting from exposure to very high doses of radiation as the result of a release. These fatalities occur days to months after exposure. Latent cancer fatalities are those resulting from the long-term effect of radiation exposure. The estimates of public health effects in this new study realistically account for the emergency planning measures in place at each reactor site, unlike some of the past studies that used generic assumptions. The results from both mitigated and unmitigated cases predict that no early fatalities will occur and average individual cancer fatality risks are very low for the unmitigated scenarios examined.

Which plants are participating in the SOARCA project?

The first phase of SOARCA analyzes examples of two major types of nuclear reactor in the United States: (1) Peach Bottom Atomic Station, a boiling water reactor (BWR) in Pennsylvania, and (2) Surry Nuclear Power Plant, a pressurized water reactor (PWR) in Virginia. After the first phase has been completed, NRC will consider whether analyses are needed for other reactor types and sites.

Does this study consider new reactors that may be built?

No. New reactor designs and containments are not part of the project. The project analyzes existing reactors.

Are terrorist acts, such as aircraft impacts, being analyzed as part of SOARCA?

No. The focus of this study is on accident scenarios—not terrorist-related ones—that could potentially lead to a radiological release into the environment. NRC addresses security-related events in separate, non-public analysis.

Are accidents at spent fuel pools considered in this study?

No. This study does not consider spent fuel pools. The project is focused on evaluating the severe and very unlikely reactor core accidents that may occur quickly at operating power reactors.

Why are the fatality numbers different from the results predicted by earlier research?

NRC is providing the most realistic, most accurate estimates calculated to date. When NRC published previous studies, the available analytical methods and data about nuclear plant operation were cruder and the results were therefore conservative. Since then, NRC and the industry have improved safety and mitigation measures in the plants. In addition, NRC has improved methods to calculate consequences. Therefore, the SOARCA project is an update to the previous research based on all the information known today.

How much different would the numbers be if NRC did the calculations the same way they were done in the past?

The purpose of the SOARCA project is somewhat different from the calculations done in the past because this project is a "best estimate" consequences analysis. In addition, NRC's knowledge, computational capabilities, and modeling methodologies are better now than in the past. A detailed report (available through Agencywide Documents Access and Management System [ADAMS]) will describe the justifications for the changes in both input values and calculation methods—regardless of their impact on the final number.

Why does NRC report individual latent cancer fatality risk and not total cancer fatalities?

Reporting the latent cancer fatality risk promotes better understanding and meaning to individuals. Cancer fatality risk provides easier comparison to other kinds of cancers and context to what the accident scenarios mean to individuals. In addition, this method better represents the risk due to proximity to the site. The focus on individuals from far away to close to the plant shows the increase in risk due to the postulated severe accident. The Environmental Protection Agency and others also commonly use cancer fatality risk as a way to report consequence.

If I live within one of the reported distances in the results of SOARCA, how do I interpret my specific risk relative to the average value reported?

The human health risks calculated in SOARCA are very small. To interpret the average individual cancer risk results from SOARCA, it is helpful to consider the NRC safety goal for cancer risk of 2 in 1 million per year. The average individual cancer risks calculated in SOARCA within the 10-mile emergency planning zone are all in the 1 in a billion to 1 in a 100 billion per year range. The average individual risk numbers decrease the further the distance out from the plant (e.g., 50 and 100 miles). The SOARCA cancer risk values are all significantly smaller than the NRC-established safety goal that "individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health."

Is this study being reviewed by outside experts?

Yes. In addition to the peer review afforded by NRC's Advisory Committee on Reactor Safeguards, an independent external peer review of scientific and technical experts will assess the methodological approach, underlying assumptions, and results obtained for Peach Bottom and Surry to ensure that they are defensible and state-of-the-art. This peer review is a common practice in research and will show both the strengths and weaknesses of the research project. NRC will continue to use the methods shown to be strengths of the research project, and the experts' comments on the weaknesses will help improve future research projects.

Enclosure 3

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

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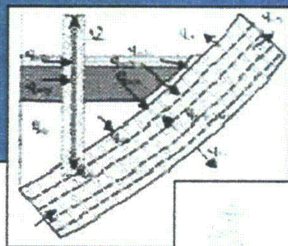

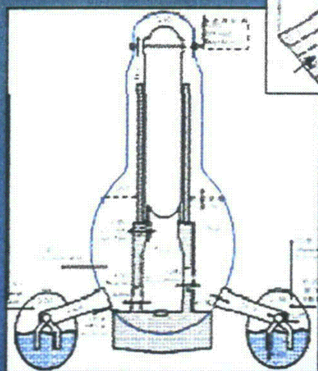

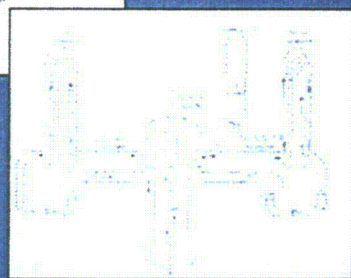

Modeling Hypothetical Accidents at Nuclear Power Plants

State-of-the-Art Reactor Consequence Analyses: Integrating research and experience about modeling accident progression, mitigation, and emergency response

Office of Nuclear Regulatory Research

Draft Documet

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The mission of the NRC is to license and regulate the Nation's civilian use of byproduct, source, and special nuclear materials in order to protect public health and safety, promote the common defense and security, and protect the environment.

Personal Statement from
Brian Sheron
Director
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
***Disclaimer-this statement is not really from office director, just an example of tone and content we might want here.*

Dear Citizen,

Much like you, we at the U. S. Nuclear Regulatory Commission care about protecting the public health and environment from the consequences of accidents at nuclear power plants. The NRC staff members have committed their professional careers to this mission. We carry out this mission in various ways, including this research project to estimate the possible public health and safety consequences in the unlikely event of a commercial nuclear power plant accident that releases radioactive material into the environment. The results of this project indicate that commercial nuclear power plants are designed and regulated to prevent accidents and to protect the public should an accident occur. We believe it validates the efforts that utilities have made over the last 25 years to improve their plant designs and operations and NRC developments in rigorous inspection methods, operator training, and emergency preparedness. All of these changes have increased overall nuclear power plant safety. I invite you to read this booklet to understand how we modeled nuclear power plant accidents using state-of-the-art methods to understand how current operation standards and regulations impact the consequences of these unlikely accidents.

KEY INSIGHTS:

- Commercial nuclear power plants are designed and regulated to prevent accidents and to protect the public should an accident occur.
- Decades of improvements to plant designs, operations, mitigation measures, and emergency preparedness have increased overall nuclear power plant safety.
- The latest analyses indicate that radiation releases could be prevented or delayed by several hours and would be substantially smaller than earlier thought because of these improvements.
- As a result, offsite consequences from severe accidents at nuclear power plants are smaller than previously predicted.
- Both mitigated and unmitigated cases predict that there will be no early fatalities.
- Average individual long-term cancer fatality risks are very low for the scenarios examined.

Acknowledgments:

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Chapter 1
Project Overview

Peach Bottom and Surry Power Stations

This chapter explains the purpose of the project and the overall process for determining the results.

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Project Overview

WHAT IS THE PROJECT'S PURPOSE?

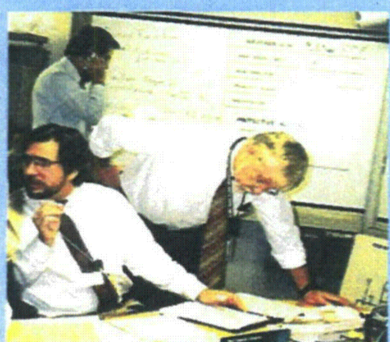
The U. S. Nuclear Regulatory Commission's (NRC's) State-of-the-Art Reactor Consequences Analyses (SOARCA) research project is designed to develop realistic estimates of the potential health effects on the public from a nuclear power plant accident, in which very unlikely scenarios could release radioactive material into the environment and potentially cause offsite health effects. The project also evaluated and improved, as appropriate, methods and models for realistically evaluating plant responses during severe accidents, including protective actions for the public (such as evacuation and sheltering), and the potential public health risk.

The NRC performed this study to obtain realistic information about the effectiveness of methods for mitigating severe accidents at nuclear power plants to prevent or minimize harm to the public.

Over the past 25 years, the NRC, industry, and international nuclear safety organizations have completed substantial research on plant response to hypothetical accidents that could damage the core and containment. That research has significantly improved the NRC's ability to analyze and predict how nuclear plant systems and operators will respond to severe accidents. During that

WHO IS THE PROJECT TEAM?

The project team consists of scientists and engineers from the NRC and the U. S. Department of Energy's National laboratories. Team members are committed to using their technical expertise to create detailed computer models to help determine realistic consequences of severe accidents at nuclear power plants. The results of this research will inform areas that the NRC and the industry should focus on to prevent and respond to severe accidents.



same time, plant owners enhanced plant designs, emergency procedures, inspection programs, and operator training, all of which have improved plant safety. Plant owners and local governments have also refined and improved emergency preparedness to further protect the public in the highly unlikely event of a severe accident.

The SOARCA team applied this accumulated research and incorporated plant enhancements to achieve a more realistic evaluation of the consequences from severe nuclear accidents. The project's results will become the foundation for communicating aspects of severe accidents and updating information from older research studies.

WHAT HAVE WE LEARNED ABOUT SEVERE ACCIDENTS?

SOARCA's results indicate that commercial nuclear power plants are designed and regulated to prevent accidents and to protect the public should an accident occur. For more than two decades, utilities have improved their plant designs and operations, mitigation measures, and emergency preparedness. All of these changes have increased overall nuclear power plant safety. Other SOARCA insights include the following:

- Accident progression is several hours slower than previously thought, resulting in a delayed release of radioactive material.
- Newly incorporated mitigative measures, as implemented according to NRC rules, can prevent radioactive releases and protect the public.
- The modeled radiation releases are delayed and relatively small.
- As a result, the individual risk of offsite consequences from severe accidents at nuclear power plants (early fatalities and long-term cancer fatalities) are much less than previously predicted.

WHAT IS THE HISTORICAL BACKGROUND OF SEVERE ACCIDENT RESEARCH?

To obtain information that helps protect the public, the NRC researched the probabilities and potential consequences of severe accidents at nuclear plants (see NUREG-xxxx for details about these past studies). Additionally, the NRC conducted several experiments testing and verifying the integrity of containment behavior. Given the limited realistic information available for these historical studies, the staff based its work on conservative assumptions about plant behavior which led to conservative results. These early research results have, at times, been misinterpreted and misused as factual data rather than estimates based on assumptions. Based

Historical Perspective: Three Mile Island and Chernobyl

Many people are familiar with the severe accidents that occurred at one U.S. and one Soviet nuclear power plant. While SOARCA did not examine these historical accidents, this type of research is aimed at obtaining information that will help prevent future accidents. Periodically, this book provides specific information about these accidents so that readers can compare the results of this study to history.


Three Mile Island

The Three Mile Island accident in Pennsylvania on March 29, 1979, resulted in major fuel damage and led to radioactive gases and contaminated cooling water filling the containment building. A very small amount of radioactivity was released, but it did not harm people, animals, or the environment.

A much more serious nuclear accident happened in 1986 at the Chernobyl power plant in the former Soviet Union. An explosion damaged the reactor core and released a very large amount of radioactive material into the environment. Several emergency responders and citizens died as a result of exposure to the material. The design of that reactor, which differed significantly from reactors operating in the U.S., made it vulnerable to such a severe accident.

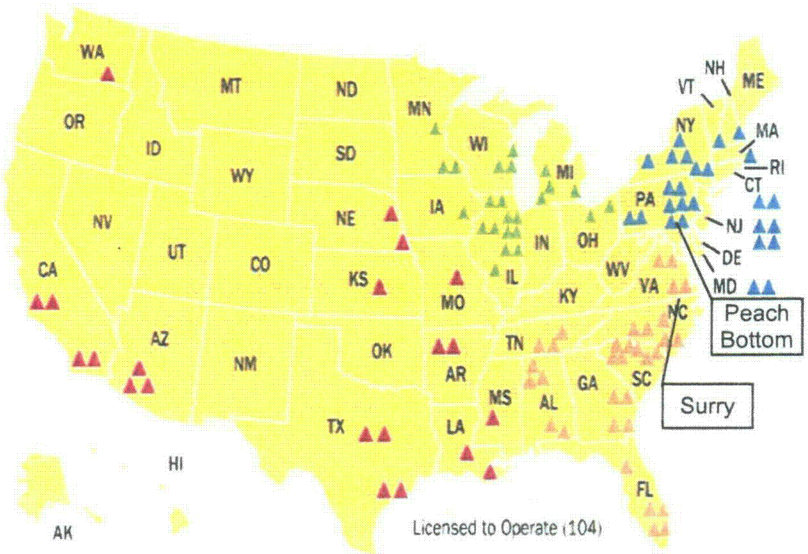

Chernobyl Site

on real-world experiments and improvements in methods and computing power, the NRC has developed better methods and information to study potential accident consequences. The SOARCA project seeks to produce more realistic and likely estimates, thereby improving public understanding of the risks and consequences of a potential accident.

WHAT IS THE PROJECT'S SCOPE?

SOARCA's first phase analyzes examples of each major type of U.S. nuclear reactor: a boiling-water reactor (BWR) and a pressurized-water reactor (PWR). The project team solicited volunteers from the nuclear industry to participate in the project. Peach Bottom (a BWR) and Surry (a PWR) were the first two sites to volunteer and they are the focus of the results of this report. The NRC's Advisory Committee on Reactor Safeguards and independent, external nuclear experts

Figure 1.1 Locations of U. S. Nuclear Reactors



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reviewed the methods and results of this first phase. Subsequent phases of SOARCA will consider analyzing other major types of reactors and include other scenarios.

HOW ARE SEVERE ACCIDENTS AND THEIR CONSEQUENCES MODELED?

The SOARCA project uses sophisticated computer models to understand how a reactor might behave under severe accident conditions and the resulting impact on the public. A computer model calculates how a system of related parts will perform under a given set of conditions. SOARCA's complex calculations are performed by powerful computers running programs designed for modeling specific accident conditions. The SOARCA project integrates information about reactor systems, components, operating history, and the impacts of emergency procedures, weather conditions, emergency planning, and population density. It uses the MELCOR computer code to model the severe accident scenarios and the MACCS2 computer code to model offsite consequences.

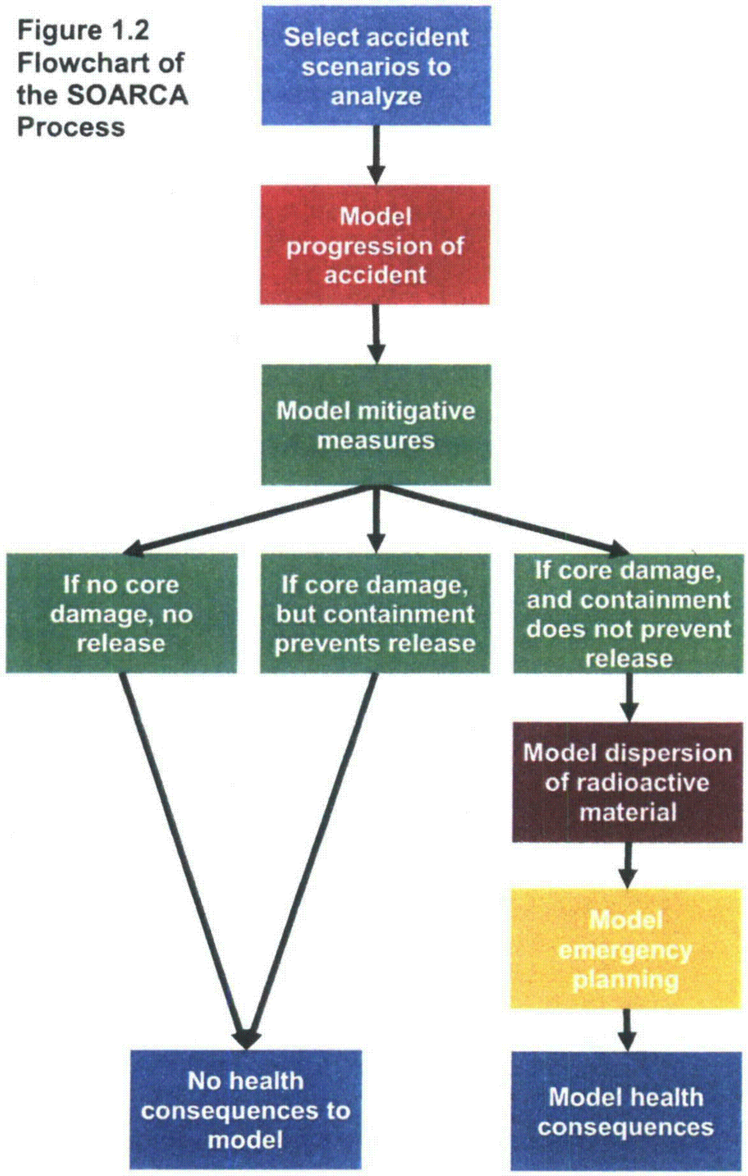
WHAT IS THE NRC RULE?

As you read through this booklet, you may learn about processes in nuclear power plants that seem hazardous. However, the NRC and industry work diligently to ensure safe operations of nuclear power plants. In support of safety, the NRC has developed many rules for the proper operation of a nuclear power plant. These rules are detailed in Title 10 of the *Code of Federal Regulations* (10 CFR). Throughout this booklet, we will refer you to some of the relevant rules so you can better understand how the NRC works to protect public health and the environment. An online version of Title 10 is available at http://www.nrc.gov/reading-rm/doc-collections/cfr/.

WHAT WERE THE STEPS OF THE PROJECT?

The SOARCA project took a step-by-step approach to analyze the potential consequences of the more likely severe accidents. The project team first determined it could learn more by rigorously and realistically quantifying a relatively few important events, rather than carrying out approximate or conservative modeling of many events. The

Figure 1.2
Flowchart of the SOARCA Process



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Control room of nuclear power plant

project then set a technical criteria to determine the scenarios on which to focus its resources (Chapter 2 describes the selection process). Then, the team ran two versions of these scenarios. For the "mitigated case" version, plant operators successfully implemented emergency plans and mitigating actions—a case that the team believes is more realistic. In order to understand the value of mitigating actions, the team also ran an "unmitigated case" scenario in which operational mitigating actions were not performed and led to a hypothetical release of radioactive material. Figure 1.2 illustrates the reasoning of this overall approach.

HOW TO USE THIS BOOKLET

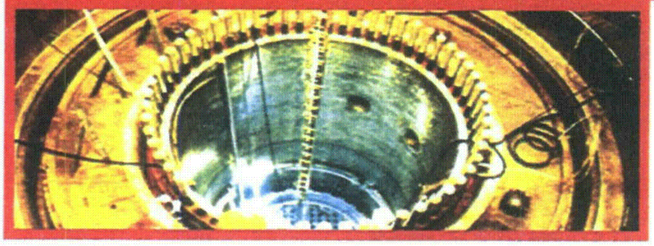
Regulators and industry groups have been researching the consequences of severe accidents at nuclear power plants since the 1950s. This booklet provides tools to help understand the processes, terminology, and results of these studies. Here are some features that you can use:

- Colored boxes like this that provide explanations of concepts
- Glossary in the appendix that defines terms
- Side boxes that provide historical information
- Side boxes that explain relevant NRC regulations
- References to information documents in the appendix

If you are viewing this online:

- Grey, underlined terms that link to the glossary in the appendix
- Grey, underlined phrases and URLs that link to the NRC Web site

Chapter 2
Progression of Accident Scenarios



Reactor Core

This chapter explains the basic information on reactor design and how accident scenarios could lead to damage of the reactor core.

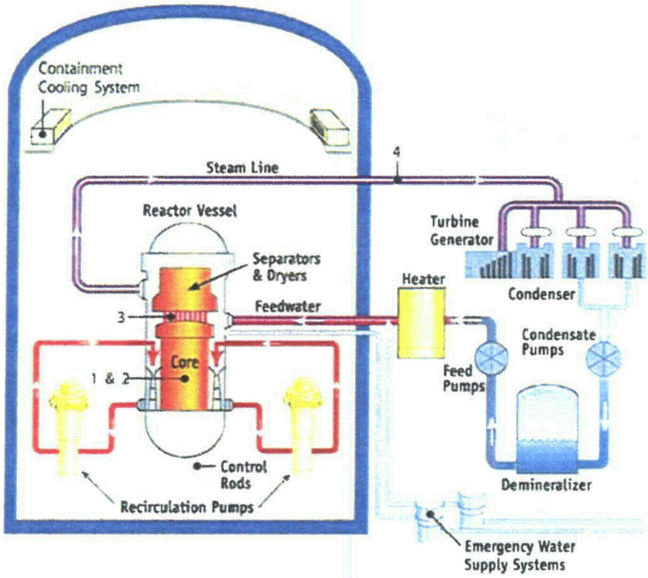
Progression of Accident Scenarios

WHAT ARE DIFFERENCES BETWEEN REACTOR TYPES?

Figures 2.1 and 2.2 describe the differences between the two U.S. commercial nuclear power plant types: boiling-water reactors (BWR) and pressurized-water

Figure 2.1 Typical Boiling-Water Reactor

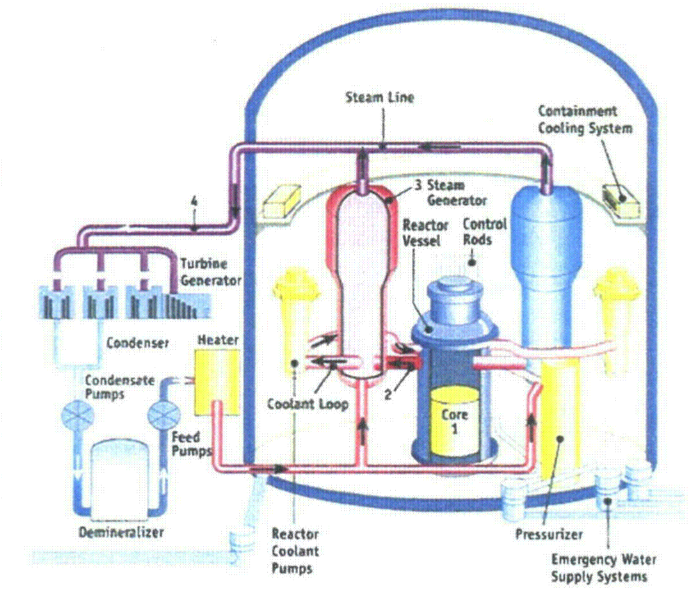
In a typical commercial BWR the following sequence occurs: (1) the reactor core creates heat, (2) a steam-water mixture is produced when very pure water (reactor coolant) moves upward through the core absorbing heat, (3) the steam water mixture leaves the top of the core and enters the two stages of moisture separation where water droplets are removed before the steam is allowed to enter the steamline, and (4) the steamline directs the steam to the main turbine causing it to turn the turbine generator, which produces electricity. The unused steam is exhausted to the condenser where it it condenser into water. The resulting water is pumped out of the condenser with a series of pumps, reheated, and pumped back to the reactor vessel. The reactor's core contains fuel assemblies which are cooled by water, which is force-circulated by electricity power pumps. Emergency cooling water is supplied by other pumps, which can be powered by onsite diesel generators. Other safety systems, such as the containment cooling system, also need electric power and can be powered by onside diesel generator BWRs contain between 370-800 fuel assemblies.



reactors (PWR). Within these two general categories, there are design variations at different sites. While both types of reactors generate electricity, their different designs may allow for different types of accidents to develop.

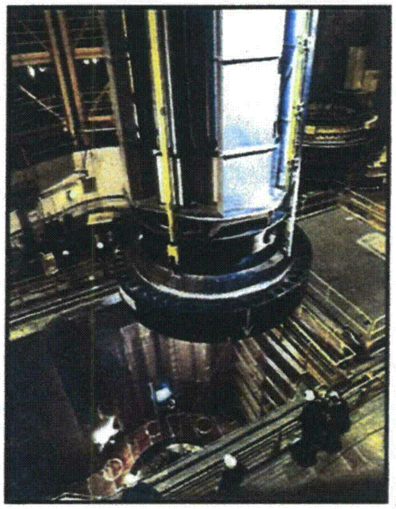
Figure 2.2 Typical Pressurized-Water Reactor

In a typical commercial PWR the following process occurs: (1) the reactor core creates heat, (2) pressurized water in the primary coolant loop carries the heat to the steam generator, (3) inside the steam generator heat from the primary coolant loop vaporizes the water in a secondary loop producing steam, and (4) the steamline directs the steam to the main turbine causing it to turn the turbine generator, which produces electricity. The unused steam is exhausted to the condenser where it is condensed into water. The resulting water is pumped out of the condenser with a series of pumps, reheated, and pumped back to the steam generator. The reactor's core contains fuel assemblies which are cooled by water, which is force-circulated by electrically powered pumps. Emergency cooling water is supplied by other pumps, which can be powered by onsite diesel generators. Other safety systems, such as the containment cooling system, also need electric power and can be powered by onside diesel generators. PWRs contain between 150-200 fuel assemblies.



Progression of Accident Scenarios

HOW WERE SCENARIOS SELECTED?

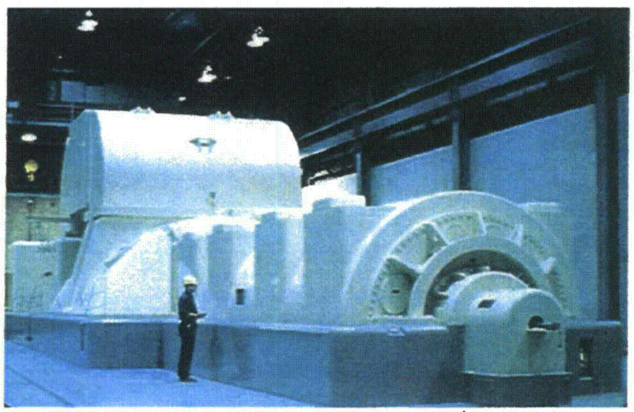


Containment area of nuclear reactor

The project team selected a set of important severe accidents to perform detailed analyses and modeling of these scenarios. This step allowed the team to focus attention and resources on the more likely accident scenarios, or groupings of incidents, that may lead to core damage at a nuclear power plant.

Core damage occurs when accident conditions heat up the reactor core to the point at which experts anticipate the fuel will be damaged. Extended core damage could lead to core melt, which is severely damaged reactor fuel that progresses to melting and movement of the core materials.

The team used site-specific probability information to determine whether an accident scenario met the threshold for consideration. This specific



Turbine at a nuclear power plant

information—data about each site's unique design, safety systems, components, and emergency plans—was computed in a probabilistic risk assessment (PRA) to determine the likelihood of the scenarios leading to core damage (also called core damage frequency). To be considered in SOARCA, an accident scenario had to have a greater than one in a million reactor years probability or greater than one in ten million reactor years probability for accidents that may bypass containment features. The team also considered scenarios that may have lower likelihoods than the threshold, but potentially higher consequences.

What is Probabilistic Risk Assessment?

The NRC assesses risk by computing the probability of an event in relationship to its consequences. A mean risk value can be represented with the equation:

Risk = Probability x Consequences

PRA involves a procedure for computing risk by asking a series of three questions called the "risk triplet":

- What can go wrong?
- How likely is it?
- What would be the consequences?

The NRC's regulatory activity uses risk information from PRAs to encourage licensees to reduce accident probabilities and to mitigate accident consequences. For this project, the SOARCA team used the information about the probabilities of certain accident scenarios to determine which ones were more important to analyze to determine their consequences. You can get more information about PRA from NRC's Web page: http://www.nrc.gov/about-nrc/regulatory/risk-informed/pra.html.

Progression of Accident Scenarios

WHAT WERE THE SOARCA SCENARIOS?

For both sites (Peach Bottom and Surry) the team modeled the following scenarios which were hypothesized to be externally initiated by seismic events.

Long-Term Station Blackout (LTSBO)—In this scenario, the station loses all alternating current power sources but battery backups are available for short-term (about 4–6 hours) operation of the safety systems.

Short-Term Station Blackout (STSBO)—In this scenario, the site loses all power, even the batteries, and therefore all of the safety systems are quickly inactivated in the "shorter term."

Additionally, the team identified two internally initiated scenarios for the PWR design at Surry. Both of these scenarios are considered "bypass events," in which radioactive materials reach the environment without having a structural containment failure.

Interfacing-System Loss-of-Coolant Accident (ISLOCA)—In this scenario, multiple check valves fail which causes a rupture in the low-pressure system outside the containment.

PWR Steam Generator

Thermally Induced Steam Generator Tube Rupture (TISGTR)—In this scenario, the reactor is performing under short term station blackout conditions, but, while the core is uncovering and heating up, extremely hot steam and hydrogen flow out into the steam generator tube. If the tube fails, radioactive material moves through a steamline, past containment and exits a relief valve.

HOW WERE THE ACCIDENTS MODELED?

The SOARCA team realistically modeled the accident scenarios and their potential for causing core damage by gathering detailed information about each site. The team gathered this information by—

- Asking plant staff for specific information about the mechanics of each plant system.
- Applying recent international research about severe accidents.
- Updating containment models based on extensive research experiments.

The team entered all of this information into a state-of-the-art computer code named MELCOR which modeled how each scenario would unfold at each

WHAT IS THE NRC RULE?
General Design Criteria

In 1971, the Atomic Energy Commission (the NRC's predecessor) published detailed design criteria for commercial U.S. power reactors which can be found in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

Criteria about quality—These criteria set requirements for how structures, systems, and components must meet standards for safety performance and establishes requirements for how licensees' quality assurance/quality control programs must be maintained.

Criteria about protection—These criteria set requirements for how commercial reactors must provide multiple layers of protection against natural phenomenon, multiple fission products, and control of the reactivity process.

Criteria about design—These criteria set requirements for reactor containment, movement of fluids, and fuel.

The NRC amends the general design criteria as necessary to reflect current research and operating experience. These criteria can be found in 10 CFR Part 50, Appendix A or online at http://www.nrc.gov/reading-rm/doc-collections/cfr/part050/

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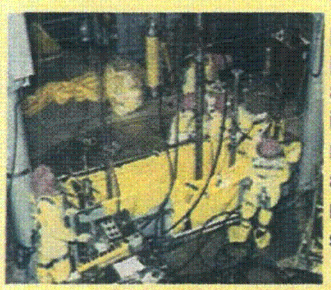
plant. The MELCOR results describe the following:

- How the plant and its emergency systems perform
- How the reactor core behaves as it heats up
- How the fuel itself, the reactor piping, and the containment building behave under extremely high heat conditions (and if the containment fails, why it fails)
- Whether radioactive material reaches the environment, and if so, how it occurs and how much material is released

Historical Perspective:
How did the Three Mile Island accident unfold?

The accident began on March 28, 1979, when the feedwater pumps stopped running due to either a mechanical or electrical failure, preventing the steam generators from removing heat. Immediately, the pressure in the nuclear portion of the plant began to increase. In order to prevent that pressure from becoming excessive, the pilot-operated relief valve opened. The valve should have closed when the pressure decreased by a certain amount, but it did not. Signals available to the operator failed to show that the valve was still open. As a result, cooling water poured out of the stuck-open valve and caused the core of the reactor to overheat to a point where about one-half of it melted. Fortunately, the vessel did not melt, the containment withstood the increased pressure, and small releases occurred through a monitored pathway. The public and environment were largely protected from the effects of the accident. While this accident resulted in an ultimately harmless radiation release, the NRC learned from this accident and imposed new regulations on the industry to increase safety. For more info: http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/3mile-isle.html



After Three Mile Island accident

This information relies on the plant's designed physical safety systems. However, nuclear plants also have a series of redundant safety measures to back up the designed safety systems. Chapter 3 discusses how the SOARCA project models the mitigating actions that can prevent release of radioactive material and ultimately protect the public.

If a scenario released radioactive material, the team used another computer code to model the release; Chapter 4 provides more details about this step.

Historical Perspective:

How did the Chernobyl accident unfold?

The accident at Chernobyl Nuclear Plant in the Ukraine (then Soviet Union) occurred on April 26, 1986. The series of events that led to this accident could not occur at U.S. commercial power reactors because of safety concerns that were documented and designed out of U.S. reactors. Chernobyl's operators decided to run an experiment under less than favorable conditions and made several mistakes while doing so. This combination of factors led to an uncontrollable and extremely rapid spike in the nuclear reaction. Within seconds, the core temperature rose above 5000 degrees Fahrenheit, melting the core and causing a steam explosion that destroyed the core and tore open the reactor building, immediately releasing large amounts of radioactive material into the air and causing several fires (some burned for 10 days). More than a dozen emergency workers died, and several fatal cases of thyroid cancer were later attributed to the accident.

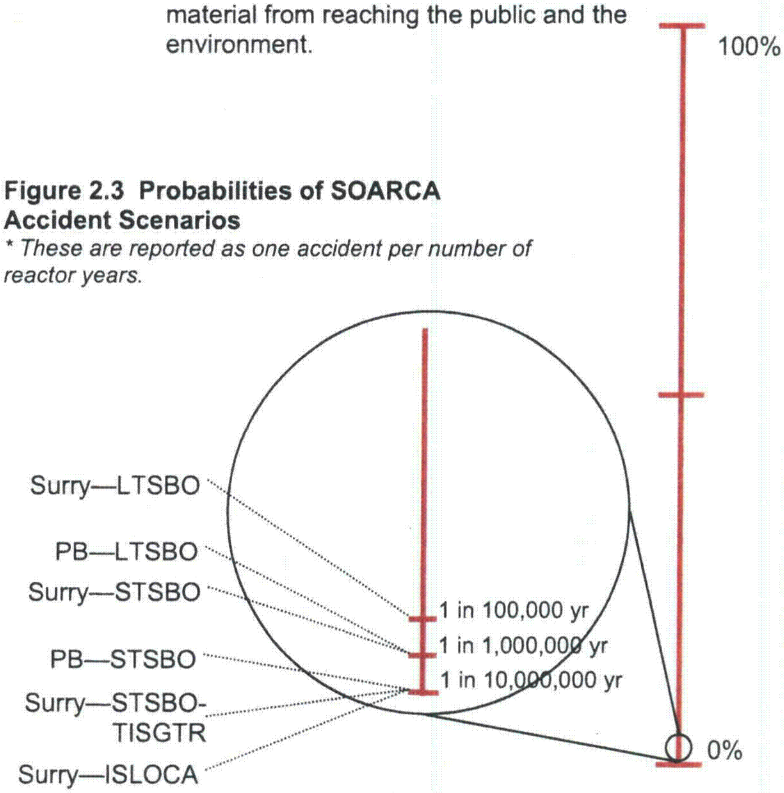
Immediately after accident at Chernobyl

HOW LIKELY ARE THESE ACCIDENTS?

Overall, the SOARCA scenarios have probabilities that range from one accident in 100,000 reactor-years to one accident per 10,000,000 reactor-years. Figure 2.3 uses a magnifying glass to demonstrate how these very small probabilities compare to likelihood of 100 percent. The chances of these scenarios ever occurring is extremely small. However, the rest of this study examines the effectiveness of actions to mitigate an accident, should one occur, and prevent radioactive material from reaching the public and the environment.

Figure 2.3 Probabilities of SOARCA Accident Scenarios

** These are reported as one accident per number of reactor years.*



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Chapter 3
Actions To Mitigate Accidents

Reactor Operating Room

This chapter explains the basic information on operator actions that mitigate the effect of accidents by preventing core damage or release of radioactive materials.

Actions to Mitigate Accidents

WHAT ARE THE MITIGATING ACTIONS?

In addition to the multiple and redundant physical systems designed to keep U.S. nuclear power plants safe, the NRC and utility companies recognize the importance of incorporating emergency measures as a backup to designed safety systems. The NRC expects that these actions will mitigate accident scenarios by preventing core damage and/or the release of radioactive material. The NRC mandates that utility companies maintain detailed emergency procedures plans for all possible accidents. These plans include the following:

Emergency operating procedures—These are detailed lists of actions for each possible nuclear power plant emergency.

Severe accident management guidelines—These are guidelines for mitigating accidents more severe than on which the reactor design is based.

Post-9/11 mitigative measures—These measures include the portable equipment nuclear plants put in place following the terrorist attacks on September 11, 2001, and are supplemented by the use of existing equipment with new procedures to run under accident conditions.

Defense-in-Depth Philosophy

The NRC's use of "defense in depth" refers to the design and operational philosophies that apply to nuclear facilities to ensure adequate public protection. This approach calls for multiple layers of protection to prevent and mitigate accidents. It includes the use of controls, multiple physical barriers to prevent release of radiation, redundant and diverse key safety functions, and emergency response measures.

HOW DOES THE NRC KNOW THESE MITIGATING ACTIONS WILL WORK?

The NRC requires its licensees to train and practice these mitigating actions in the plant simulators that are present at each site. The agency also verifies that the licensees have implemented the post-9/11 measures to ensure that they have proper equipment, procedures, and training.

SOARCA is the first detailed modeling and quantification of the value of these mitigating actions. The project accomplished this by running two cases of each scenario: a mitigated case and an unmitigated case. This illustrates the value of these actions for mitigating an accident, preventing release, and ultimately protecting the public.

Historical Perspective:
How has security improved since 9/11?

The NRC increased its requirements for nuclear power plants to protect against sabotage, terrorism, and other intentional attacks. The new requirements include:
-increased patrols
-additional security posts
-additional physical barriers
-vehicle checks at further distance
-enhanced coordination with law enforcement and military
-more restrictive site access controls

For more info: www.nrc.gov/security/faq-911.html#2

HOW ARE MITIGATING ACTIONS MODELED?

For each plant, two cases of each scenario are modeled.

Mitigated Case—In the first case, the SOARCA team modeled what would happen if the operators successfully executed the mitigating actions. The team gathered information from staff at each site about how long it would take to implement the mitigating actions. The MELCOR calculations included this information to see how the mitigating actions would impact accident progression. If operators successfully execute these procedures,

Mitigating Actions

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Historical Perspective:
How did emergency operations improve after Three Mile Island?

Since the Three Mile Island accident, the NRC imposed additional training requirements. The nuclear industry now runs its reactor operators through emergency situations using full-scale control room simulators. The NRC emergency operations center is continually manned and ready to respond if something happens. The NRC has inspectors at each plant and requires more information from plant owners so we can ensure they make the correct safety decisions.

then consequences to the public will be prevented or minimized.

Unmitigated Case—In the second case, the team modeled what would happen if the operational mitigating actions were not executed. These cases modeled the sequence of events that lead to core melt, release of radioactive materials, and consequences to the public.

WHAT IS THE TIMING OF MITIGATING ACTIONS?

Detailed MELCOR modeling demonstrated that plant operators had enough time during accident scenarios to perform the necessary mitigating actions. In the unmitigated cases, the SOARCA team did not include this information in the modeling calculations. Therefore, since these accidents led to a release, the team modeled the release, emergency response, and health consequences. Figure 3.1 illustrates the timeline for the Peach Bottom long term station blackout scenario from the blackout until the release starts (in an unmitigated situation) and compares that with the mitigating actions timeline.

Figure 3.1 Timing of Accident Progression

Peach Bottom Long Term Station Blackout

Mitigated Scenario	Hrs	Unmitigated Scenario
Station blackout	0:00	Station blackout
Operators position, connect and start portable electricity Operators manually control coolant (by 4 hours)		
Operators align and start portable pumps (4 to 10 hours)	4:00	Backup batteries deplete
	5:00	Reactor coolant flow stops
Accident Mitigated No Release	10:00	
	15:00	Water has boiled off
	20:00	Lower head and containment fail Release starts

Mitigating Actions

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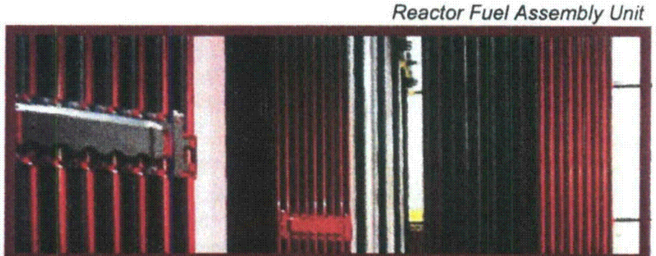
Reactor Fuel Assembly Unit



Chapter 4
Release of Radioactive Material

This chapter explains how the project modeled the release of radioactive material and what information is used in the calculations.

Release of Radioactive Material

The SOARCA models showed the mitigated cases to prevent or minimize core damage and therefore prevent a release of radioactive material. To determine the extent of the value of the mitigative actions, the team modeled the unmitigated scenarios which did lead to a hypothetical release. This step of the project models how the radioactive material would disperse from the site through the environment and population following core damage. The MELCOR computer code models the behavior of radioactive materials to the point that they exit containment. From this point, the SOARCA team used the MACCS2 computer code to model the dispersion of the radioactive material and situations of human exposure (Chapters 5 and 6 cover this in more detail). This chapter describes the process by which SOARCA models radioactive material as it exits the containment and enter the environment.

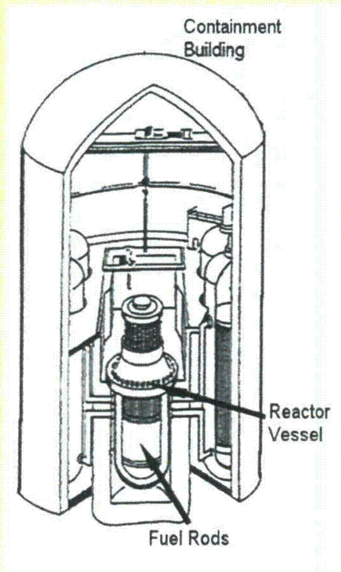
How does containment work?

As part of the defense-in-depth philosophy, the NRC requires all currently operating reactors to have these three layers of containment that protect the public and environment from potential releases of radioactive material:

Fuel Rods—long, slender tubes that hold fissionable material (fuel) for nuclear reactor use. Fuel rods are assembled into bundles which are loaded individually into the reactor core.

Reactor Vessel—metal enclosure that holds the reactor core and the coolant.

Containment Building—enclosure around a nuclear reactor to confine fission products that otherwise might be released to the atmosphere in the event of an accident.



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WHICH RADIONUCLIDES DOES SOARCA MODEL?


Fuel Rods in Bundles

In the MELCOR code, SOARCA considers decay heat from approximately 800 radioactive substances, or radionuclides, that are produced during the accident progression and organized by chemical group. From this first set, the offsite consequences computer code (MACCS2) tracks approximately 60 radionuclides based on the length of their half-life, their biological importance, and amount produced in the fission process.

Cesium and Iodine—These two radionuclide groups are important for offsite consequence analysis because they can interact readily with the human body and lead to significant radiation doses.

Noble gases—These radionuclides (such as krypton and xenon) are chemically very stable and unlikely to move easily into human exposure situations, even though they are readily released after an accident.

Other radionuclides—MACCS2 models several other radionuclides in the core. However, they receive less attention in the discussion of the results because their releases are slower and smaller.

Modeling Release

WHAT INFORMATION IS INCLUDED IN MODELING?

- How physical and chemical processes influence the behavior of radioactive material while the core heats up.
- How the extremely high temperatures influence particles' behavior at the molecular level and their physical states (e.g., gas or aerosolized particles).
- How the radioactive material moves within the containment and reactor coolant system (before exiting containment).
- How engineered safety systems (such as sprays and fan coolers) impact the behavior of radioactive material to prevent release.
- When radioactive material exits containment.

HOW ARE RADIOACTIVE MATERIALS MODELED TO EXIT CONTAINMENT?

Peach Bottom Scenarios

Long-Term Station Blackout— 20 hours after the scenario begins, molten core material fails the bottom head of the reactor vessel, pours onto the containment floor, spreads across the floor, and contacts the steel containment shell melting a hole through it.

Short-Term Station Blackout— 8 hours after the scenario begins, molten core material fails the bottom head of the reactor vessel, pours onto the containment floor, spreads across the floor, and contacts the steel containment shell melting a hole through it.

Surry Scenarios

Long-Term Station Blackout— 45 hours after the scenario begins, the pressure in the containment building reaches the building's ultimate failure pressure resulting in tearing of the containment liner and cracking of the reinforced concrete.

Short-Term Station Blackout— 25 hours after the scenario begins, the pressure in the containment building reaches the building's ultimate failure pressure resulting in tearing of the containment liner and cracking of the reinforced concrete.

Interfacing-System Loss-of-Coolant Accident— The scenario begins with the hypothesized random failure of 2 valves in series that ruptures a pipe outside of the containment building, providing a path from the reactor core to the environment which bypasses containment. About 10 hours after the scenario begins, the accident progresses to the point where aerosolized radioactive particles are released through this path.

Short-Term Station Blackout Thermally Induced Steam Generator Tube Rupture— 3.5 hours after the scenario begins, high-pressure high-temperature gas circulating through the reactor coolant system causes a steam generator tube rupture allowing aerosolized radioactive particles to flow out of the broken tube bypassing the containment building.

Figure 4.1 Percentage of Radioactive Material Released from Reactor Core after 24 Hours

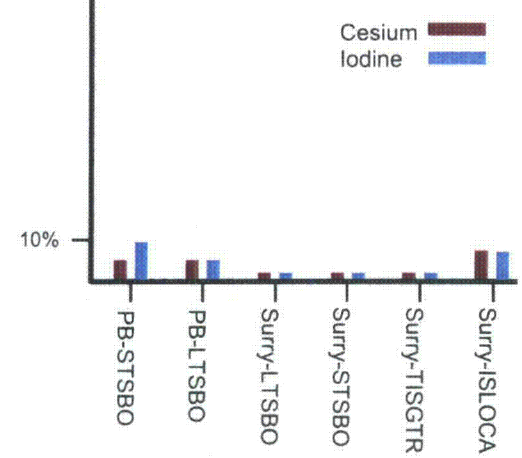
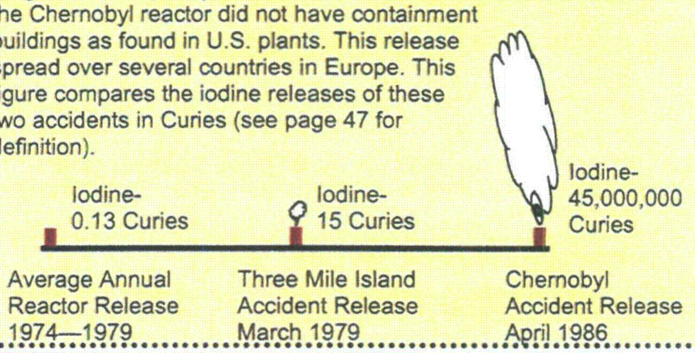


Figure 4.1 compares the percentage of the reactor core that each accident scenario releases. Note that these percentages are very small—they are less than 10 percent of the radioactive materials in the core at the beginning of the accident. Also note that it takes up to 24 hours to release this small amount of radioactive material.

Historical Perspective:

What were releases after Three Mile Island and Chernobyl?

Since it had a strong containment, the accident at Three Mile Island resulted in a very small release of radioactive material and no health or environmental consequences resulted from this accident. The Chernobyl release was much more severe because the accident progressed so quickly and released so much radioactive material and the Chernobyl reactor did not have containment buildings as found in U.S. plants. This release spread over several countries in Europe. This figure compares the iodine releases of these two accidents in Curies (see page 47 for definition).



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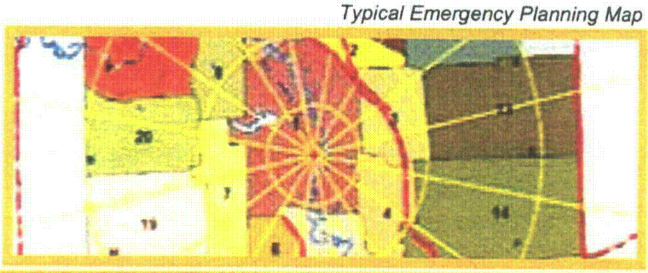
Typical Emergency Planning Map



Chapter 5
Modeling Emergency Plans

This chapter explains the requirements of emergency planning and how these were modeled.

Modeling Emergency Plans

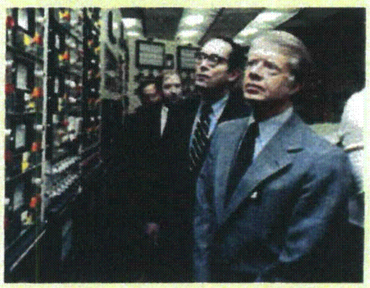
The mitigated cases were able to prevent a release. Therefore, for these scenarios, even if the community had to evacuate there would not be health consequences from the release. To determine the extent of the value of the mitigating actions, the team modeled the unmitigated scenarios that could lead to a release and necessitate an evacuation. The computer modeling of the dispersion of radioactive material includes the modeling of emergency plans. The NRC requires each site to have detailed emergency plans outlining how onsite personnel would work to prevent a release in the case of an accident and, if a release were to occur, how offsite personnel would coordinate evacuation and sheltering of surrounding populations. The computer code uses this information to model the evacuation of the public in the case of a severe accident.

In nearly all scenarios, the delayed timing of the accidents (even without mitigative actions) allowed sufficient time for local populations to evacuate safely. This chapter provides more information about how the SOARCA project modeled emergency plans.

Historical Perspective:

What about the emergency plans at Three Mile Island?

The Three Mile Island accident revealed the need for better coordination between nuclear power plant operators and Federal, State, and local government emergency response organizations. Following the accident, the NRC's emergency preparedness regulations were changed to require each nuclear power plant owner to submit the radiological emergency response plans of State and local governments for the plume exposure pathway and the ingestion exposure pathway emergency planning zones (EPZs).



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WHAT IS EMERGENCY PLANNING?

The NRC requires nuclear power plants to have onsite and offsite emergency plans as a defense-in-depth measure. The NRC provides support to emergency planning by ensuring that the licensee can execute its plans and coordinating the State and Federal responses. Emergency plans focus on protecting public health and safety with the following major objectives:

1st Objective—Fix the plant. To accomplish this, the NRC requires the utilities to have onsite response that includes technical, physical, and management staff that can respond within the first hour of the start of the accident. Each year, the licensee trains and drills this capability and the NRC inspects it.,

2nd Objective—Evacuate and shelter the local populations if the plant cannot be fixed. To accomplish this, the NRC requires utilities to have offsite response support from the county and State agencies. The Federal Emergency Management Agency inspects this capability every two years. Emergency planning zones (EPZs) help define where detailed protective action strategies would be used during an emergency. Every plant must have emergency action levels approved by the NRC that dictate when to declare and emergency well before core melt or radiation release. This timing is designed to ensured that the plan will be implemented before the plant is in a serious state and that members of the public are well on their way to evacuation before the release.

INTERORGANIZATION COOPERATION

The NRC requires each nuclear power plant to have an emergency response center from which the utility coordinates local, State, and Federal responses to emergencies. Additionally, the NRC has a 24/7 emergency response center that provides support during emergencies.

Emergency Planning

Modeling Emergency Plans

WHAT INFORMATION IS INCLUDED IN EMERGENCY PLAN MODELING?

In the mitigated cases, the modeling results indicated that there would not be a release. Therefore the SOARCA team did not model emergency plans. However, for the unmitigated comparison cases, the SOARCA team modeled the specific emergency plans for each site. The team used detailed information that included the following:

- 2005 population distributions around each site
- Actual evacuation time estimates from emergency plans

Using information from each site's emergency plans, the SOARCA team created evacuation groups, or cohorts, and modeled their evacuation timing in conjunction with the timing of the accident and its release. See Table 5.1 for more information.

WHAT ARE EMERGENCY PLANNING ZONES?

Two EPZs around each nuclear power plant help define what protective action strategies will be used during an emergency. Predetermined protective action plans are in place for the EPZs and are designed to avoid or reduce dose from potential exposure of radioactive materials. Utilities base the size and shape of their EPZs on site-specific conditions, unique geographical features of the area, and demographic information.

Plume Exposure Pathway EPZ—The plume exposure pathway EPZ has a radius of about 10 miles from the reactor site. The actions for this EPZ include sheltering, evacuation, and the use of potassium iodide where appropriate.

Ingestion Exposure Pathway EPZ—The ingestion exposure pathway EPZ has a radius of about 50 miles from the reactor site. The actions for this EPZ include a ban of contaminated food and water.

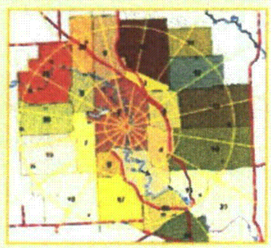

10 Mile EPZ Map

Table 5.1 Evacuation Cohorts

Cohort 1	(1) school populations within the 0—10 miles of the site and (2) shadow population including those members of the public that evacuate from an area that has not been ordered to evacuate
Cohort 2	the public within 0—10 miles of the site
Cohort 3	(1) special needs population including residents of hospitals, nursing homes, assisted living communities, and prisons within 0—10 miles of site and (2) the tail which is the 10 percent of the public from 0—10 miles of site who are assumed to take longer to prepare for evacuation

WHAT DOES MODELING DEMONSTRATE ABOUT EMERGENCY PLANNING?

The MACCS2 computer code models public evacuation, sheltering, and the return of the population after the event. Figure 5.2 illustrates the results of the timing of the accident, release, and evacuation. Since accident scenarios take several hours to result in core melt and a release, this generally provides time for a large portion of the population to evacuate before radiation exposure.

WHAT IS THE NRC RULE?

According to 10 CFR 50.47 "Emergency Plans" the licensees must demonstrate to the NRC that they will take adequate protective measures in the event of a radiological emergency. The offsite emergency plans must establish procedures for the licensee to notify State and local response organizations and provide early notification and clear instruction to the public in the plume exposure pathway. For details see http://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-0047.html.

Modeling Emergency Plans

WHAT DOES MODELING DEMONSTRATE ABOUT EMERGENCY PLANNING?

Figure 5.2 illustrates the modeled timing of the Peach Bottom long-term station blackout scenario and the timing of emergency response. The different colors of the bars indicate the length of time that each cohort continues normal activity, shelters from radiation exposure, and evacuates from the area. The bottom of the graph notes key accident progression and emergency response events. In each modeled scenario, the plants follow their stated emergency response plans about when to activate their emergency notification systems (sirens) and direct the public to evacuate. The Figure 5.2 demonstrates how cohort groups are sheltered and evacuated before radioactive release begins.

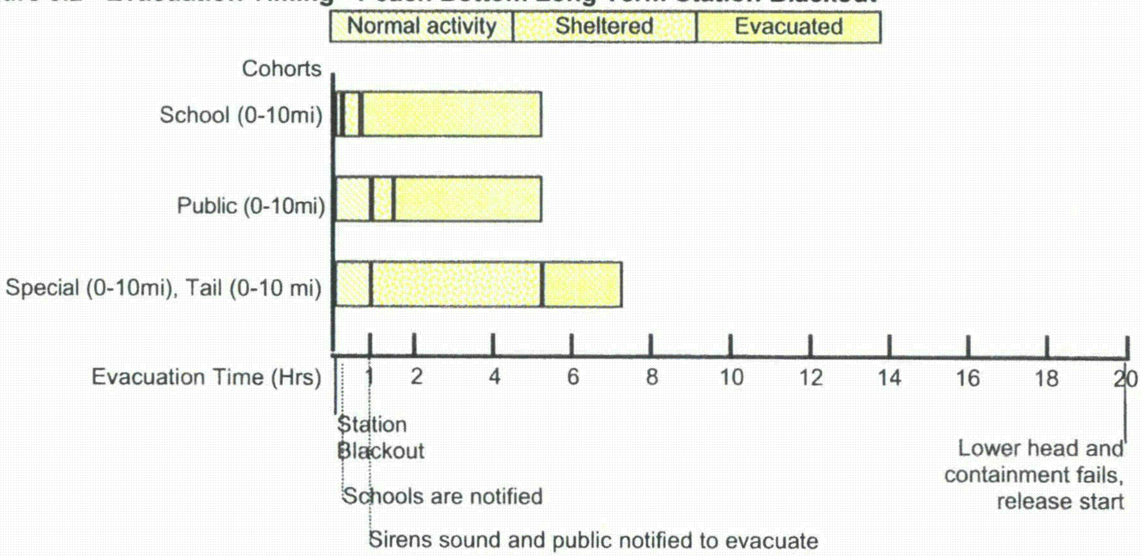
How does the NRC enforce preparedness at nuclear reactors?

Emergency response center

The NRC ensures that the personnel at nuclear power plants know what to do in the case of an emergency by—

- Requiring licensed operators to train on emergency operating procedures yearly in the operating room simulators
- Running regular emergency drills
- Making technical support available from NRC Headquarters 24 hours, 7 days a week

Figure 5.2 - Evacuation Timing - Peach Bottom Long Term Station Blackout



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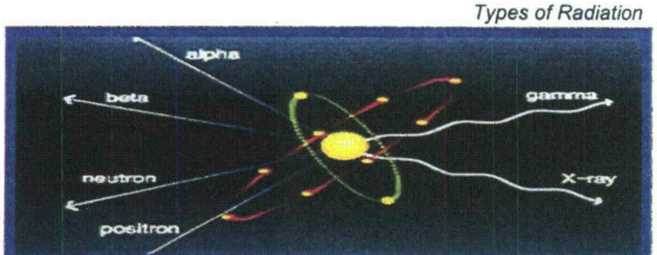
Types of Radiation

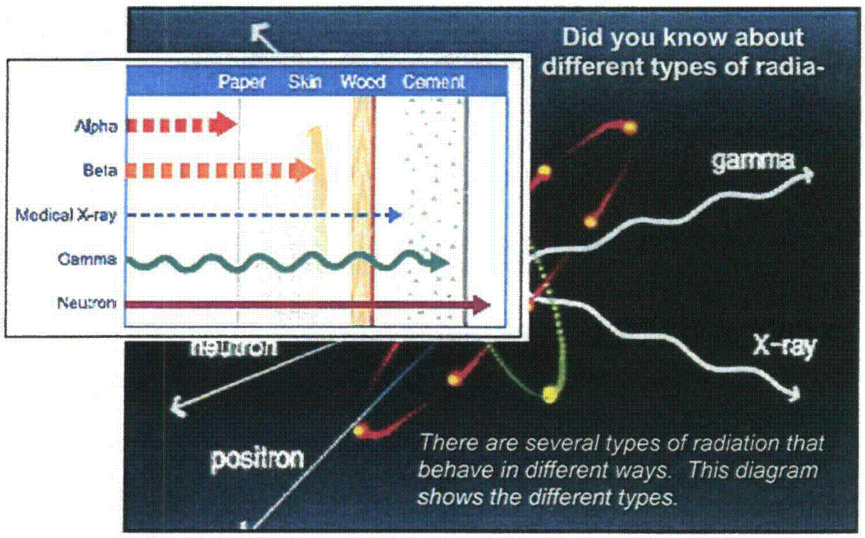


Chapter 6
Modeling Health Effects

This chapter describes the models to calculate health consequences and the results based on the SOARCA project.

Health Effects

In the mitigated cases, the mitigating actions were able to prevent a release, providing evidence of the value of these actions for protecting the public health. To determine the extent of the value of the mitigative actions, the team modeled the unmitigated scenarios which did lead to a release and subsequent health consequences. However, even in these scenarios, modeling indicated that there would not be any early fatalities (because of the length of time for the accident progression) and the possibility of long-term cancer fatalities was very small (because of slowly developing and relatively small releases). These results indicate that commercial nuclear power plants are designed and regulated to prevent accidents and to protect the public should an accident occur. Even in situations in which operators unsuccessfully enact emergency procedures, the risk of consequences to the public are extremely small. This chapter provides an explanation and background information about how SOARCA modeled the health consequences.



There are several types of radiation that behave in different ways. This diagram shows the different types.

HOW ARE HEALTH CONSEQUENCES REPORTED?

Early Fatalities—Human fatalities that occur shortly after exposure to large doses of radiation (usually within a few months). The report expresses this number as the average individual likelihood of an individual fatality.

Long-Term Cancer Fatality Risk—The cancer fatality risk that occurs years after exposure to radiation. The report expresses this number as dependent on whether a given scenario occurs and represents the average individual cancer mortality risk due to radiation exposure following the hypothesized accident.

How is radiation measured?

Units that measure how much radioactive material decays over a period of time:
Curie (Ci)
Becquerel (Bq)

Units that measure the effects of ionizing radiation on humans:
rem
Sievert (Sv)

A Geiger counter is a tool that measures radiation in the environment.

WHAT'S THE RULE?

Radiation dose limits for emergency responders to a nuclear accident:
5 rem—Any activity
10 rem—Protecting valuable property
25 rem—Lifesaving or protecting large population

Set in U. S. Environmental Agency protection action guides—the NRC regulates within these guidelines: http://www.epa.gov/radiation/rert/pags.html

HOW ARE LONG-TERM CANCER FATALITY RISKS MODELED?

The NRC is committed to using state-of-the-art science and professional judgment to determine long-term cancer effects in the case of a severe reactor accident. Modeling long-term cancer fatality risk has been a very controversial issue because of inconclusive human

Health Effects

evidence regarding risk at low doses. To provide additional information how different calculation assumptions may affect the potential range of health consequences, the SOARCA project developed long-term cancer risk estimates following the assumptions of two leading theoretical positions. The full SOARCA report (NUREG-xxxx) presents each of these results for three distance intervals:)1) 0—10 miles, (2) 0—50 miles, (3) 0—100 miles. The two theoretical positions include the following:

Linear-no-threshold model—This model suggests that any amount of radiation exposure (no matter how small) can incrementally lead to a dose that may result in cancer. It is a basic assumption used in many regulatory limits (e.g., occupational exposure).

Truncation models—These models suggest that below certain doses, one cannot quantify a cancer risk due to uncertainty. By using this value, SOARCA provides information about long-term cancer health effects relative to the doses that people receive in different exposure groups. SOARCA uses three dose truncation values:

10 mrem per year—represents a small dose

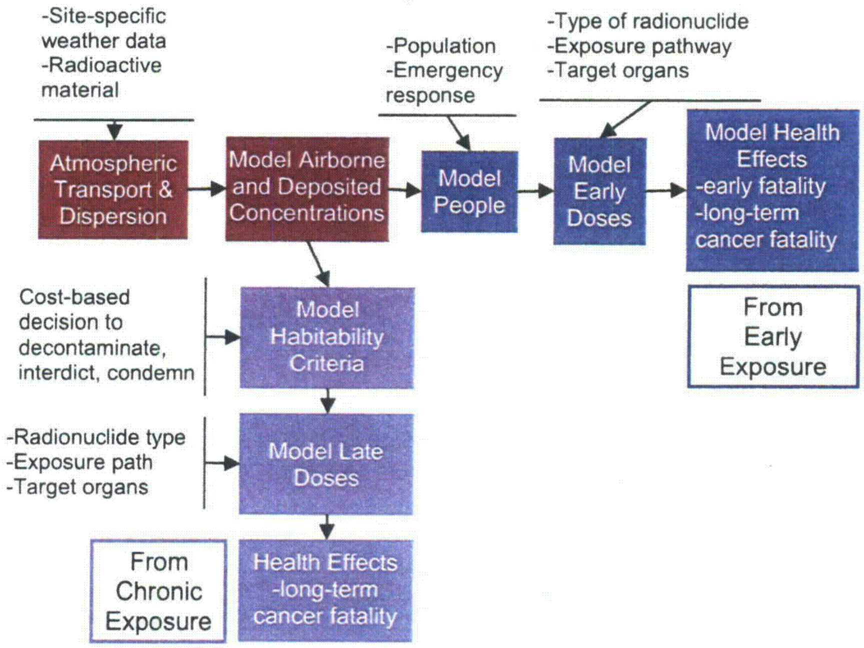
360 mrem per year—represents radiation levels that naturally occur in the environment

5 rem per year with a 10 rem lifetime cap—represents the Health Physics Society Position Statement

WHAT INFORMATION IS IN THE MODELING?

Figure 6.1 illustrates the relationships among the information that the computer model (MACCS2) calculates to determine health effects from radiation exposure after a severe accident. The model uses weather pattern information to model how the radioactive material may disperse through the environment. It then models early exposure and chronic exposure.

Figure 6.1 Modeling Health Effects



Early Exposure—This calculation determines how many long-term cancer fatalities and early fatalities result from early exposure. In this model, SOARCA determines how many people will be exposed based on the population and the timing of the emergency response (remember from <u>Chapter 5</u> that most people can be evacuated before release in most scenarios). Once SOARCA estimates how many people might be exposed, it uses information about how each type of radionuclide will impact the body based on the characteristics of the radionuclide and path of exposure (inhalation or full body exposure).

Late Chronic Exposure—This calculation models long-term cancer fatalities that result from exposure people receive after they are permitted to return

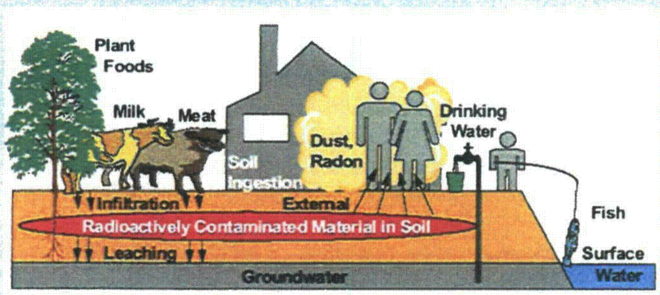
Health Effects

home. For the Surry model, SOARCA uses habitability criteria from the U.S. Environmental Protection Agency "Manual of Protective Action Guides for Nuclear Incidents" to determine when the population can return to the area. For the Peach Bottom model, SOARCA uses Pennsylvania-specific habitability criteria. Once they return home, people may become exposed to remaining amounts of radiation, and SOARCA uses information about how each type of radionuclide will impact the body based on the characteristics of the radionuclide and path of exposure (inhalation or full body exposure).

Historical Perspective:

What about all the thyroid cancer after Chernobyl?

After Chernobyl, thousands of children were exposed to radioactive iodine which, over the years, has resulted in a high incidence of thyroid cancer. These children became exposed to radioactive iodine mainly by eating contaminated foods. Although, tragically, several have died, many have been effectively treated so the death rate is low. The SOARCA project does not include the food pathway because the dietary habits of the U.S. population are different. In essence, SOARCA assumes that if a severe nuclear reactor accident contaminated an environment, the population would be instructed not to eat the food and would have access to enough food from other areas that they would not need to consume the contaminated food—hence averting radioiodine exposure through ingestion.



Pathways of human exposure to radiation

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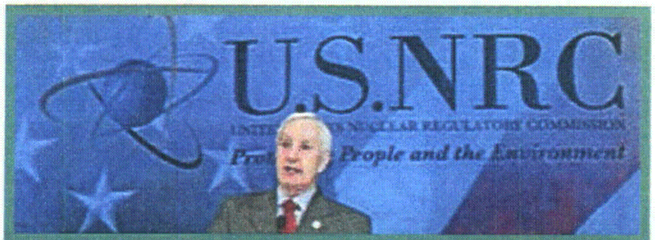
This chapter summarizes the results and conclusions about the research project.

Chairman Klein of NRC



Chapter 7
Results and Conclusions

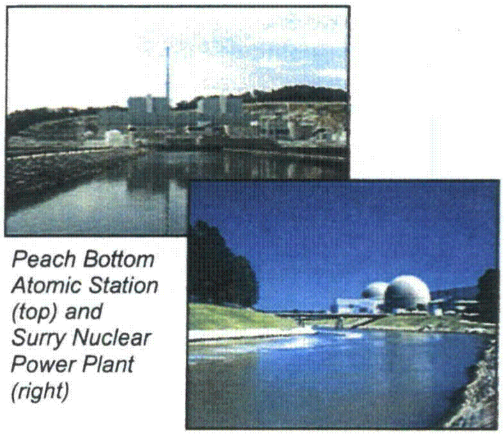
Results and Conclusions

The mitigated scenarios demonstrate that reactor designs, operator actions, and regulations can prevent consequences to the public if a severe accident were to occur. However, for the benefit of comparison, the SOARCA team ran hypothetical scenarios that demonstrate the consequences if operational mitigating actions were not performed. The results of the unmitigated scenarios are presented in this chapter and demonstrate that even in these scenarios the consequences to the public are very low.

HOW DO MITIGATED SCENARIOS UNFOLD?

Early Fatalities—In the modeling for each of these scenarios, the operators were able to prevent core damage. Since the accidents in these scenarios were effectively mitigated, there is no release of radioactive materials resulting in early fatalities.

Long-Term Cancer Fatalities—In the modeling for each of these scenarios, the operators were able to prevent core damage. Since the accidents in these scenarios were effectively mitigated, there is no release of radioactive materials and therefore no risk of long-term cancer fatalities.



Peach Bottom Atomic Station (top) and Surry Nuclear Power Plant (right)

Which corrective actions mitigate accidents?

The operators mitigated the accidents by using measures implemented following the events of September 11, 2001. In the blackout scenarios, they supplemented the use of existing equipment with new procedures to run under blackout conditions.

HOW DO UNMITIGATED SCENARIOS UNFOLD?

Early Fatalities—There are no early fatalities for the unmitigated scenarios. Even though these scenarios did lead to core damage, the release of radioactive material occurs after long periods of time which allow for protective actions of the population (including evacuation, sheltering, and relocation). Therefore, in these situations, no one is initially exposed to large amounts of radioactive material.

Long-Term Cancer Fatalities—In these scenarios, the average individual risk of a long-term cancer fatality is modeled to be very small—regardless of which distance interval or calculation model is used. Tables 7.1 and 7.2 summarize the modeling for the scenarios at each plant.

Table 7.1 Peach Bottom Unmitigated Scenarios

	LTSBO	STSBO
How likely is the accident to occur?	1 event in one million years	1 event in 10 million reactor years
From the initiating event, how long until the release happens?	20 hours	8 hours
How much of the core is released after 48 hours?	Iodine - 4%; Cesium - 2%	Iodine - 11%; Cesium - 2%
What is the risk —the annual average individual risk of a long-term cancer fatality for this scenario within 10 miles of the plant?	8 in 10 billion	7 in 100 billion
What does this risk number mean?	The risk numbers are a very small fraction of the NRC's reactor safety goal for cancer risk of 2 in 1,000,000 per year	

Conclusion

Results and Conclusions

The following table summarizes the modeling of the Surry unmitigated scenarios.

Table 7.2 Surry Unmitigated Scenarios

	LTSBO	STSBO
How likely is the accident to occur?	1 event in 100,000 years	1 event in 1,000,000 reactor years
From the initiating event, how long until the release happens?	45.5 hours	25.5 hours
How much of the core is released after 48 hours?	Iodine - <1%; Cesium - <1%	Iodine - 1%; Cesium - <1%
What is the risk—the annual average individual risk of a long-term cancer fatality for this scenario within 10 miles of the plant?	7 in 10 billion	1 in 10 billion
What does this risk number mean?	The risk numbers are a very small fraction of the NRC's reactor safety goal for cancer risk of 2 in 1,000,000 per year	

Table 7.2 Surry Unmitigated Scenarios (continued)

	ISLOCA	TISGTR
How likely is the accident to occur?	1 event in 10 million years	1 event in 10 million reactor years
From the initiating event, how long until the release happens?	10 hours	3.5 hours
How much of the core is released after 48 hours?	Iodine - 9%; Cesium - 9%	Iodine - 1%; Cesium - <1%
What is the risk—average individual risk of a long-term cancer fatality per year for this scenario?	2 in 10 billion	1 in 10 billion
What does this risk number mean?	The risk numbers are a very small fraction of the NRC's safety goal for cancer risk of 2 in 1,000,000 per year	

HOW DOES THE NRC DETERMINE THE VALIDITY OF THIS STUDY?

Peer Review —As part of a research process, peer reviews identify the strengths and weaknesses of a research project. Independent, external experts in the field of risk analysis and severe accident research reviewed the SOARCA process and results. The SOARCA team incorporated this feedback into the final published document.

Uncertainty—Scientific research strives for strong validity based on using high-quality data and reasonable assumptions. However, since the best data are not always available, researchers run tests of "uncertainty" to specifically identify the weaknesses in data and assumptions. This step is an important part of making research "transparent." SOARCA used a statistical Monte Carlo approach to identify uncertainties.

WHAT HAVE WE LEARNED ABOUT SEVERE ACCIDENTS?

The results of this project indicate that commercial nuclear power plants are designed and regulated to prevent accidents and to protect the public should an accident occur. Over two decades of improvements in plant designs, operations, mitigation measures, and emergency preparedness have increased overall nuclear power plant safety. Other SOARCA insights include the following:

- Accident progression is several hours slower than previously thought, resulting in a delayed release of radioactive material.
- Newly incorporated mitigative measures, as implemented according to NRC rules, can protect the public from radioactive releases.
- The modeled radioactive releases are delayed and relatively small.
- As a result, offsite consequences from severe accidents at nuclear power plants are much smaller than previously thought.

WHAT ABOUT THE FUTURE OF COMMERCIAL NUCLEAR POWER?

The NRC conducted the first phase of the SOARCA project during 2006—2009, at a time when there was much public discourse about a "renaissance" of commercial nuclear power. However, the mission of the NRC is to license and regulate the commercial nuclear power plants in order "to protect public health and safety, promote the common defense and security, and protect the environment." This means that, rather than take a position to promote the use of nuclear power, the NRC must ensure that the nuclear power plants operate safely. Under this framework of safety, the NRC conducted the SOARCA research project to better understand the impact of decades of improved research, operations, and regulation on the consequences of hypothesized accident scenarios. The results of the SOARCA project validate efforts that utilities have made to improve their plant designs and operations and NRC developments in rigorous inspection methods, operator training, and emergency preparedness. Over the last 25 years, all of these changes has increased overall nuclear power plant safety, and the staff at the NRC will continue to diligently perform its responsibilities to protect the public regardless of the political views about commercial nuclear power

OPENNESS

As a Federal agency committed to serving the public, the NRC operates transparently with respect for differing views of its stakeholders and the public. The results and methods of this research project are of great interest to many public citizens. Therefore, the SOARCA team worked diligently to make the research methods of this project transparent and the results comprehensible in the technical report (NUREG-xxxx), which is publicly available.

~~DRAFT DOCUMENT~~

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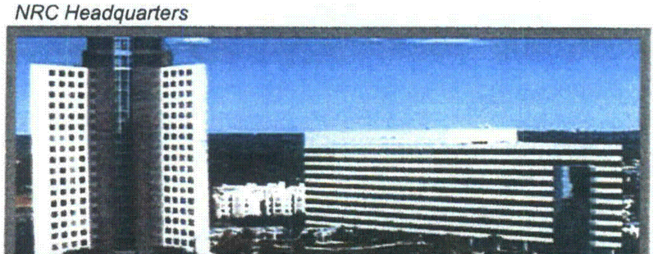
NRC Headquarters



Appendix

Glossary

References and Resources

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GLOSSARY

Advisory Committee on Reactor Safeguards (ACRS)—As described in the Atomic Energy Act of 1954, as amended, the ACRS is an internal review committee that reviews and advises the Commission with regard to the licensing and operation of production and utilization facilities and related safety issues, the adequacy of proposed reactor safety standards, technical and policy issues related to the licensing of evolutionary and passive plant designs, and other matters referred to it by the Commission.

Boiling-Water Reactor—In a typical boiling-water reactor the reactor core creates heat and a single loop both delivers steam to the turbine and returns water to the reactor core to cool it. The cooling water is force-circulated by electrically powered pumps. Emergency cooling water is supplied by other pumps, which can be powered by onsite diesel generators. Other safety systems, such as the containment building air coolers, also need electric power.

Coolant—A substance circulated through a nuclear reactor to remove or transfer heat. The most commonly used coolant in the United States is water. Other coolants include heavy water, air, carbon dioxide, helium, liquid sodium, and a sodium-potassium alloy.

Core Damage—Events leading to heat up of the reactor core to the point at which severe fuel damage is anticipated or uncovery and heat up of the reactor core to the point at which prolonged oxidation and severe fuel damage are leading to release of radioactive material from the fuel.

Core Damage Frequency—An expression of the likelihood that, given the way a reactor is designed and operated, an accident could cause the fuel in the reactor to be damaged.

Early Fatalities—Human fatalities that occur shortly after exposure to radiation, usually within a few weeks.

Feedwater—Water supplied to the reactor pressure vessel (in a boiling-water reactor or the steam generator (in a pressurized-water reactor) that removes heat from the reactor fuel rods by boiling and becoming steam. The steam becomes the driving force for the plant turbine generator.

Long-Term Cancer Fatalities—Cancer fatalities that occur years after exposure to radiation.

MACCS2 Code—A general purpose tool for estimating offsite impacts following release of radioactive material. MACCS2 is applicable to diverse reactor and nonreactor situations. It considers atmospheric transport and dispersion under time variable meteorology, short- and long- term mitigation actions, and exposure pathways to determine stochastic health effects and economic costs.

MELCOR Code—An integrated, engineering-level computer code used to model the progression of postulated accidents in light-water reactors as well as nonreactor systems (e.g., spent fuel pool and dry cask). MELCOR is a modular code consisting of three general types of packages: (1) basic physical phenomena, (2) reactor-specific phenomena, and (3) support functions. These packages model the major systems of a nuclear power plant and their associated interactions.

Mitigating Actions—Actions designed to mitigate accident scenarios by preventing core damage and/or the release of radioactive material.

Pressurized-Water Reactor—In a typical commercial pressurized light-water reactor (1) the reactor core creates heat, (2) pressurized water in the primary coolant loop carries the heat to the steam generator, and (3) the steam generator vaporizes the water in a secondary loop to drive the turbine, which produces electricity.

Radiation—Alpha particles, beta particles, gamma rays, x-rays, neutrons, high-speed electrons, high-speed protons, and other particles capable of producing ions. Radiation, as used in 10 CFR Part 20 "Standards for Protection Against Radiation," does not include nonionizing radiation, such as radio waves or microwaves, or visible, infrared, or ultraviolet light (see also 10 CFR 20.1003, "Definitions").

Reactor Fuel—Fissionable material that nuclear reactors use. It is held in long slender tubes called fuel rods and bundled into fuel assemblies before insertion into the reactor.

More term definitions are available online at the NRC Glossary at www.nrc.gov/reading-rm/basic-ref/glossary.html

Appendix

REFERENCES

U.S. Nuclear Regulatory Commission. State-of-the-Art Reactor Consequences Analysis Report. NUREG-XXXX. 200X.

U.S. Nuclear Regulatory Commission. State-of-the-Art Reactor Consequence Analysis—Reporting Offsite Health Consequences. SECY-08-0029. March 4, 2008.

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