U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

MODELING POTENTIAL REACTOR ACCIDENT CONSEQUENCES

State-of-the-Art Reactor Consequence Analyses: Using decades of research and experience to model accident progression, mitigation, emergency response, and health effects

FOREWORD

The U.S. Nuclear Regulatory Commission's (NRC's) Office of Nuclear Regulatory Research recommends and conducts research necessary for licensing and other regulatory functions of the NRC. Our focus is on nuclear safety and security of nuclear reactors, other nuclear facilities, and radioactive materials. We partner within the NRC, and with Federal agencies, industry research organizations, and international counterparts and organizations to conduct these activities.

We conducted the State-of-the-Art Reactor Consequence Analyses (SOARCA) project to estimate the potential consequences from unlikely accidents involving a commercial nuclear power plant that could release significant quantities of radioactive material into the environment. This project first modeled a set of accident scenarios for two nuclear power plants, Peach Bottom and Surry, which represent two of the most common types of plants operating in the United States. We recently completed an additional study of the Sequoyah nuclear power plant, which is representative of another common plant type in the U.S., as well as an uncertainty analysis for the three plants. SOARCA considers plant design and operational changes not reflected in earlier assessments. The project also considers NRC's development of rigorous oversight processes and use of operating experience along with improvements in operator training and emergency preparedness. In addition, we've improved the analytical tools that NRC used to perform SOARCA based on decades of national and international research.

One of SOARCA's objectives is to improve communications about hypothetical accident scenarios with stakeholders. Stakeholders include members of the public along with Federal, State, and local authorities, academia, citizen groups, and the companies that operate nuclear power plants. We have documented the SOARCA results in a series of reports that include NUREG-1935, "State-of-the-Art Reactor Consequence Analyses Main Report," and NUREG/CR-7110, Volume 1, "Peach Bottom Integrated Analyses," and Volume 2, "Surry Integrated Analyses," as well as NUREG/CR-7245, "Sequoyah Integrated Deterministic and Uncertainty Analyses" and reports on uncertainty analyses. Because the NUREG reports rely on highly technical explanations, we developed this brochure as a plain-language summary of SOARCA's methods, results, and conclusions. We invite you to read this brochure about how we used state-of-the-art methods to model these unlikely nuclear power plant accidents to understand their potential impacts on public health and safety.

KEY RESULTS:

- When operators are successful in using onsite equipment during the accidents analyzed in SOARCA, they can prevent the reactor from melting, or delay or reduce releases of radioactive material to the environment.
- SOARCA analyses indicate that all modeled accident scenarios, even if operators are unsuccessful in stopping the accident, progress more slowly and release smaller amounts of radioactive material than calculated in earlier studies.
- As a result, public health consequences from severe nuclear power plant accidents modeled in SOARCA are smaller than previously calculated.
- The delayed releases calculated provide more time for emergency response actions such as evacuating or sheltering for affected populations. For the scenarios analyzed, SOARCA shows that emergency response programs, if implemented as planned and practiced, reduce the risk of public health consequences.
- Both mitigated (operator actions are successful) and unmitigated (operator actions are unsuccessful) cases of all modeled severe accident scenarios in SOARCA cause essentially no risk of death during or shortly after the accident.
- SOARCA's calculated longer term cancer fatality risks for the accident scenarios analyzed are millions of times lower than the general U.S. cancer fatality risk.

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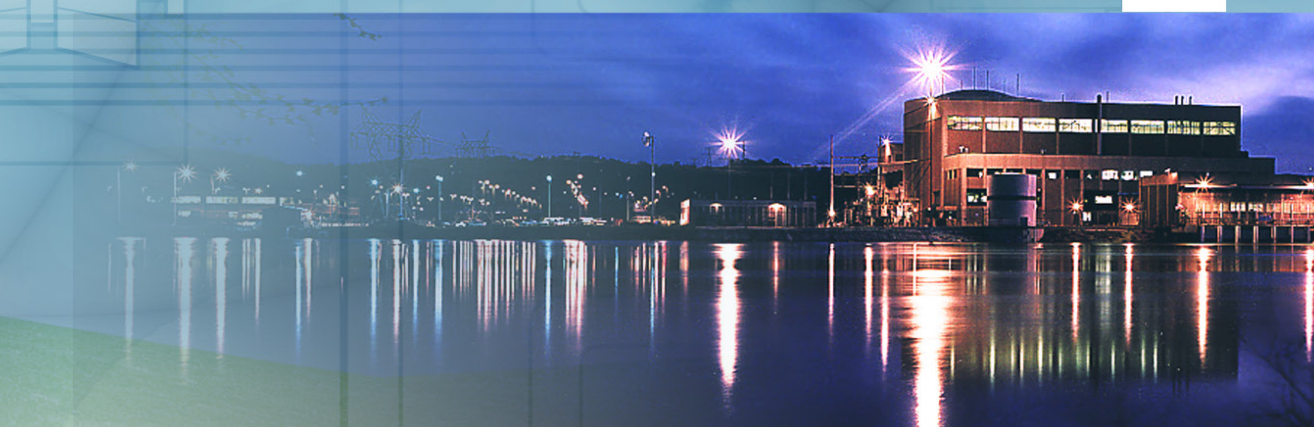
ACRONYMS

ac	alternating current
BWR	boiling water reactor
CDF	core damage frequency
CFR	Code of Federal Regulations
CRAC	Calculation of Reactor Accident Consequences
dc	direct current
EPA	U.S. Environmental Protection Agency
EPZ	emergency planning zone
ETE	evacuation time estimate
HPS	Health Physics Society
ISLOCA	interfacing systems loss-of-coolant accident
LCF	long-term cancer fatality
LNT	linear no-threshold
LOCA	loss-of-cooling accident
LTSBO	long-term station blackout
LWR	light-water reactor
MACCS	MELCOR Accident Consequence Code System
NRC	U.S. Nuclear Regulatory Commission
PRA	probabilistic risk assessment
psi	pounds per square inch (pressure)
PWR	pressurized-water reactor
SBO	station blackout
SGTR	steam generator tube rupture
SOARCA	State-of-the-Art Reactor Consequence Analyses
SST	siting source term
STSBO	short-term station blackout

PROJECT OVERVIEW

CHAPTER 1



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

How to Use this Brochure

This brochure provides tools to help understand SOARCA's processes, terminology, and results. Here are some features that you can use:

- Colored side boxes such as this one explain concepts, provide historical information, or explain relevant NRC regulations.
- Glossary in the appendix defines terms.
- References in the appendix provide a list of information documents.

If you are viewing this online:

- Gray, underlined phrases and URLs are linked to the NRC Web site.

Who Is the Project Team?

The project team included engineers and scientists from NRC and two contractors, Sandia National Laboratories and dycoda, LLC. The team's expertise included probabilistic risk assessment, heat transfer and fluid flow, emergency response, atmospheric dispersion, and radiation health effects. Team members focused their technical expertise on creating and applying detailed computer models to help determine realistic consequences of severe nuclear power plant accidents.

WHAT IS THE RESEARCH PROJECT'S PURPOSE?

The U.S. Nuclear Regulatory Commission's (NRC's) State-of-the-Art Reactor Consequence Analyses (SOARCA) research project calculated the realistic outcomes of severe nuclear power plant accidents that could release radioactive material into the environment. The computer models that produced these calculations incorporated decades of research into reactor accidents as well as the current design and operation of nuclear power plants. To provide perspective between SOARCA results and the more conservative estimates of severe reactor accident outcomes found in earlier NRC publications, SOARCA results are compared to the results of one of these previous publications: NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," commonly referred to as the 1982 Siting Study. The SOARCA report and this brochure help NRC to communicate severe-accident-related aspects of nuclear safety to you, the public; Federal, State, and local authorities; and nuclear power plant licensees. The SOARCA project also supports NRC initiatives such as resolving certain lessons learned from the Fukushima accident in Japan.

HOW IS SOARCA STATE-OF-THE-ART?

NRC considers SOARCA a state-of-the-art project because (1) it models accidents with the latest plant-specific and site-specific information, (2) it uses an improved understanding of how radioactive material behaves during an accident, (3) it examines emergency response comprehensively, and (4) it combines modern computer-modeling capabilities and detailed computerized plant models.

NRC, the nuclear power industry, and domestic and international nuclear safety organizations have extensively researched plant response to potential accidents that could damage the reactor fuel and the containment building, which is designed to keep radioactive material from reaching the environment. This research has significantly improved NRC's ability to develop computer models of how nuclear plant systems and operators would respond to severe accidents. When NRC developed the SOARCA plant models, the staff interviewed plant personnel and examined current plant equipment configurations to incorporate each facility's most current design and operational information. This updated information includes:

- Plant owners improved plant safety through enhanced plant designs, emergency procedures, inspection programs, and operator training

- Plant owners have also increased power production (referred to as "power uprates") and lengthened operating times between replacing used fuel in the reactor – these actions changed the types and amounts of radioactive material in used reactor fuel.
- Plant owners improved severe accident mitigation strategies, including NRC-required enhancements made after the terrorist attacks of September 11, 2001, to respond to fires and explosions. These mitigating enhancements [10 CFR50.54 (hh)(2)] also help mitigate the events triggered by natural occurrences such as an earthquake.
- Plant owners and local governments have refined and improved emergency preparedness programs and equipment to further protect the public in the unlikely event of a severe accident.

All of these changes have been considered in SOARCA. The SOARCA team applied this accumulated research and incorporated plant changes to more realistically evaluate the potential health consequences from severe nuclear reactor accidents.

HOW DOES SOARCA DIFFER FROM PAST SEVERE ACCIDENT STUDIES?

NRC has previously estimated the probabilities and potential health consequences of severe accidents and documented this research in reports such as WASH-1400, "Reactor Safety Study. An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants", NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants", and NUREG/CR-2239. The SOARCA Report, NUREG-1935, contains details about some of these past studies. Since the publication of the earlier studies, NRC has participated in many severe accident research programs. This work has improved our understanding of how heat is transferred and radioactive material moves through reactor systems during severe accidents and how radioactive material might get out of the containment building and move through the surrounding environment. NRC incorporated these research results into SOARCA's computer codes. In addition, the SOARCA study used a more complete and detailed computer model of the reactor, containment, and other buildings onsite. Because SOARCA is based on decades of research and uses improved modeling tools, the study generates more realistic results than past efforts such as the 1982 Siting Study. These past studies were based on then-existing plant descriptions and knowledge of how severe accidents would occur. However, we now know that the predictions from these past studies are out of date for realistically understanding severe accident consequences.

What Is a Severe Accident?

A severe accident is a type of accident that may challenge safety systems at a level much higher than expected and can cause substantial damage to the reactor core.

A reactor accident occurs when the plant cooling water systems are no longer removing heat from the reactor fuel (the "core" of the reactor). The fuel rods, when overheated during a severe accident, can also react with steam and release hydrogen which can escape from the reactor vessel and accumulate in the containment and reactor building. Extensive core damage could melt reactor fuel, which would settle at the bottom of the reactor vessel that is designed to hold the fuel. The reactor vessel is surrounded by the containment building. If cooling water is not restored, and the accident progresses further, the melted fuel could rupture the bottom of the reactor vessel, with the melted fuel flowing onto the containment floor. Radioactive material would be released from the fuel into the containment atmosphere and could potentially escape containment if there were any available leakage paths. A severe accident may involve hydrogen burns and can cause failures of containment buildings, unless mitigated by operator actions.

HOW ARE SEVERE ACCIDENTS AND POTENTIAL HEALTH CONSEQUENCES MODELED?

The SOARCA project used specialized computer programs to calculate the effect a severe accident could have on an operating nuclear reactor and the possible impact on the public. These programs integrate information about reactor systems, components,

What Are NRC Regulations?

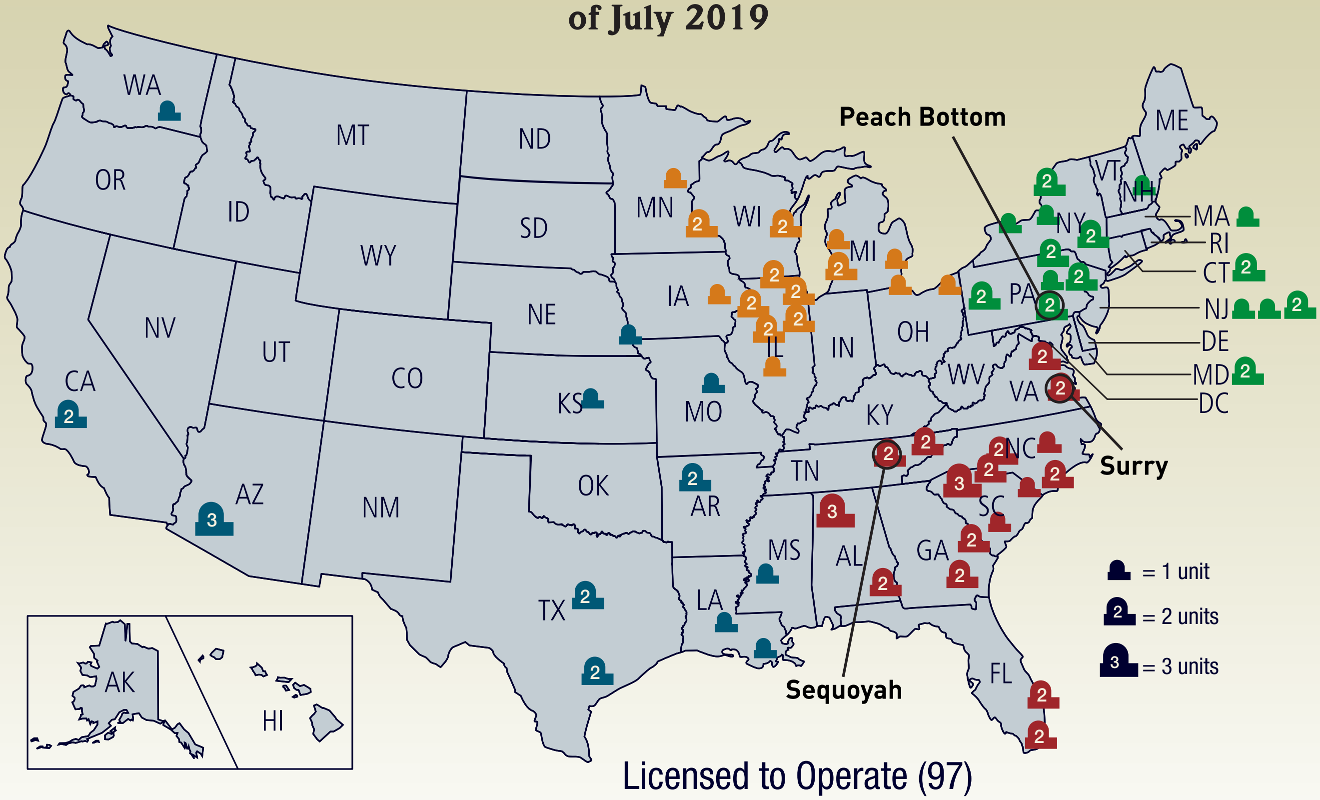
NRC works to ensure safe operation of nuclear power plants, by developing rules for the proper design, construction, and operation of a nuclear power plant. These rules are detailed in Title 10 of the *Code of Federal Regulations* (10 CFR). Throughout this brochure, we will refer you to some of the relevant rules so you can better understand how NRC works to protect public health and the environment. An online version of 10 CFR is available at http://www.nrc.gov/reading-rm/doc-collections/cfr/.

operating history, and the impacts of emergency procedures, weather conditions, emergency planning, evacuation time estimates, and population.

WHAT WERE THE STEPS OF THE PROJECT?

The SOARCA project took a step-by-step approach to calculate the potential consequences of the analyzed severe accidents. Considering the availability of resources for conducting the study, the team decided to perform highly detailed analyses of a small number of important accident scenarios. Therefore, the team selected a threshold to help select scenarios to analyze (Chapter 2 of this brochure describes the selection process). SOARCA aimed to assess the benefits of of mitigation measures put in place after 9-11 for responding to fires and explosions. In the Sequoyah plant analysis the staff considered additional equipment added to the plants as result of Fukushima events. We also wanted to provide a basis for comparison to past analyses of severe accident scenarios before these mitigation measures existed. The project therefore analyzed the selected scenarios twice: first assuming that the event proceeds without the 10 CFR 50.54(hh) mitigation measures, called "unmitigated" and then assuming that the 10 CFR 50.54(hh) mitigation is successful, called "mitigated". For scenarios leading to an offsite

Figure 1.1 Location of All Operating Reactors in The United States as of July 2019



release of radioactive material, SOARCA then analyzed the material's atmospheric dispersion, the surrounding area's emergency response, and potential health consequences. Figure 1.2 illustrates this overall approach.

HOW DOES NRC DETERMINE THE VALIDITY OF THIS STUDY?

Peer Review— A peer review is a review of a research project by experts not involved in the project. These experts examine the methods and results of the research and help improve the work by identifying the project's strengths and weaknesses. The SOARCA team assembled a panel of independent, external experts in the fields of risk analysis, severe accident research, emergency preparedness, and radiation health effects. For the first two plants, Peach Bottom and Surry, this group reviewed SOARCA's methodology, underlying assumptions, results, and conclusions to ensure that they are technically sound and state-of-the-art. For the same reasons, NRC's Advisory Committee on Reactor Safeguards (ACRS, a standing group of nuclear safety experts) also reviewed the Peach Bottom and Surry analyses and provided comments. The SOARCA team has incorporated the experts' feedback into the reports. NRC's ACRS will also review the Sequoyah analyses.

Sensitivity and Uncertainty Analyses—Scientific research strives for valid results based on high-quality data and reasonable assumptions. Because data can be sparse and uncertain, however, researchers work systematically to identify any weaknesses in data and assumptions and to consider alternatives. This step is an important part of making research results transparent and understandable. NRC staff used sensitivity analyses to compare how varying individual input assumptions affect the outcomes. The results of these sensitivity analyses show that the SOARCA results are reasonable considering known uncertainties. In addition, NRC took a systematic look at potential sources of uncertainty and their impact on SOARCA results. The uncertainty analyses use a statistical approach to assess the uncertainties in a more integrated fashion (see Chapter 7).

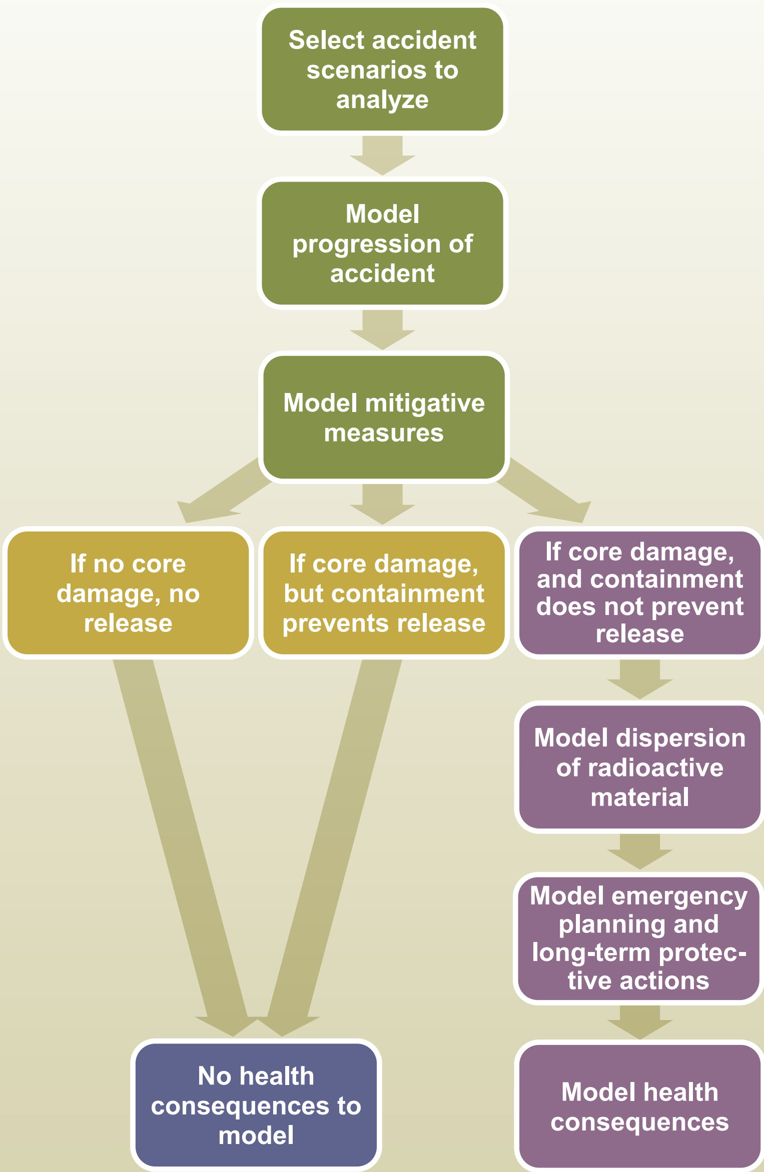
What Computer Codes Were Used for SOARCA?

SOARCA uses two specialized computer codes to analyze severe accidents and offsite consequences. The first, MELCOR, calculates accident timing and event progression using plant design information and models for the accident phenomena. The second, MACCS (MELCOR Accident Consequence Code System), calculates the offsite health consequences of an airborne release of radioactive material using site-specific information for the area and radiological release data from MELCOR.

The MELCOR code was peer reviewed in 1991 by experts from national laboratories, universities, and MELCOR code users. This peer review provided an independent assessment of the technical adequacy of the code. The peer reviewers' recommendations were incorporated into NRC's research and development plan for the code, which has also been checked, or "validated", against numerous experimental results over the past several decades.

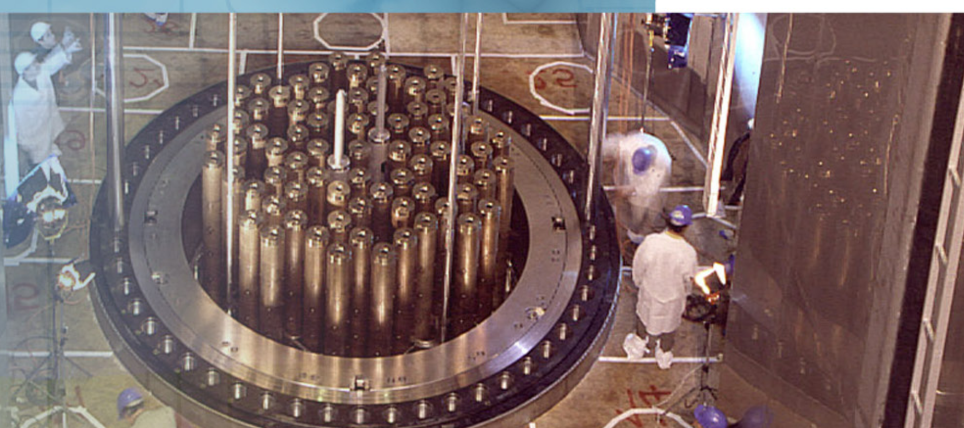
An expert panel review of the MACCS code and SOARCA's MACCS modeling choices was conducted in August 2006, prior to the start of specific work as part of the Peach Bottom and Surry analyses. This expert panel review and the NRC staff recommendations influenced much of the development that has been undertaken specifically to support SOARCA.

Figure 1.2 Flow Chart of the SOARCA Process



PROGRESSION OF ACCIDENT SCENARIOS

CHAPTER 2



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

WHICH PLANTS DID SOARCA STUDY?

SOARCA analyzed three of the most common types of operating U.S. nuclear plants. These are the Peach Bottom Atomic Power Station in Pennsylvania, the Surry Power Station in Virginia and the Sequoyah Nuclear Power Plant in Tennessee. Peach Bottom is a General Electric- designed BWR with a Mark I containment. Surry is a Westinghouse-designed PWR with a large dry containment, and Sequoyah is a Westinghouse-designed PWR with an ice condenser containment.

These three plants, depicted in Figure 2.1, also were part of earlier studies.

WHAT ARE THE DIFFERENCES BETWEEN REACTOR TYPES?

Figures 2.2, 2.3, and 2.4 describe some of the major differences between BWRs and PWRs. Within these two general types of U.S. commercial nuclear reactors, many variations exist in the design of systems, components, and containments at different sites. Figure 2.4 further describes the unique features of an ice condenser containment.

Figure 2.1 Peach Bottom (top), Surry (Middle), and Sequoyah (bottom)

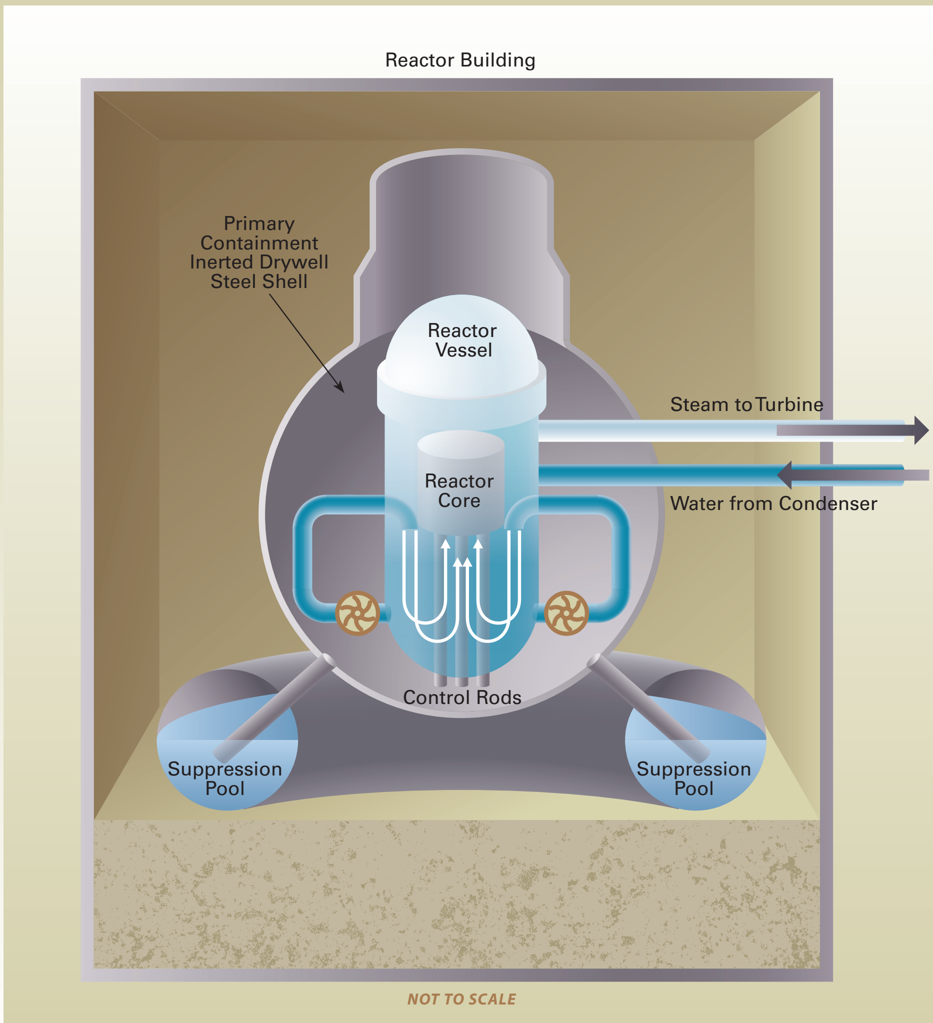
HOW WERE SCENARIOS SELECTED?

The project team sought to focus its attention and resources on the important severe accident scenarios for Peach Bottom, Surry, and Sequoyah found in past risk studies, such as NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants". The project narrowed its approach by using an accident sequence's possibility of damaging reactor fuel (also called the reactor "core"), or core damage frequency (CDF), as an indicator of risk.

The SOARCA scenarios were selected from the results of existing probabilistic risk assessments (PRAs). The scenario-selection process used updated and benchmarked standardized plant analysis risk (SPAR) models and available plant-specific information from 2005 for Peach Bottom and Surry, and 2014 for Sequoyah. Core damage scenarios from previous staff and licensee PRAs were identified and combined into common core damage groups that have similar timing and response for important severe accident phenomena and similar containment or safety systems. The groups were screened according to their approximate contributions to CDFs to identify the most risk-significant groups. SOARCA analyzed scenarios with a CDF equal to or greater

Figure 2.2 Typical U.S. Boiling Water Reactor with Mark I Containment

A BWR cools the reactor core and generates steam to turn a turbine using a single loop of water, as distinct from a PWR (see figure 2.3) that has separate loops for cooling the reactor and generating steam. Heat from nuclear fission in the reactor core converts the water to steam. The steam travels through the steam line to the turbine generator where it turns the generator to make electricity. The steam then enters the condenser where it is cooled back into liquid water and is pumped back into the reactor to repeat the process. The BWR's water is pressurized to about 1,100 pounds per square inch (psi) pressure so it boils at about 550 °F. A typical BWR core contains between 400 and 800 fuel assemblies, and each fuel assembly holds 75 to 100 fuel rods. The BWR in this figure is shown with a Mark I style of containment. More information is available at http://www.nrc.gov/reactors/bwrs.html



What is a Probabilistic Risk Assessment?

Probabilistic risk assessment (PRA) is an engineering approach to systematically identify potential nuclear power plant accident scenarios and estimate their likelihoods of occurrence and consequences. Each accident scenario begins with an initiating event (such as a loss of offsite power or earthquake) followed by a combination of equipment failures and operator actions that can lead to core damage and the release of radioactive materials from the containment. The information developed by a PRA is useful in identifying plant vulnerabilities. Pioneered by NRC in the 1970s, PRA has been adopted by nuclear power plant operators and regulators worldwide as a tool that complements other approaches to assess nuclear power plant safety.

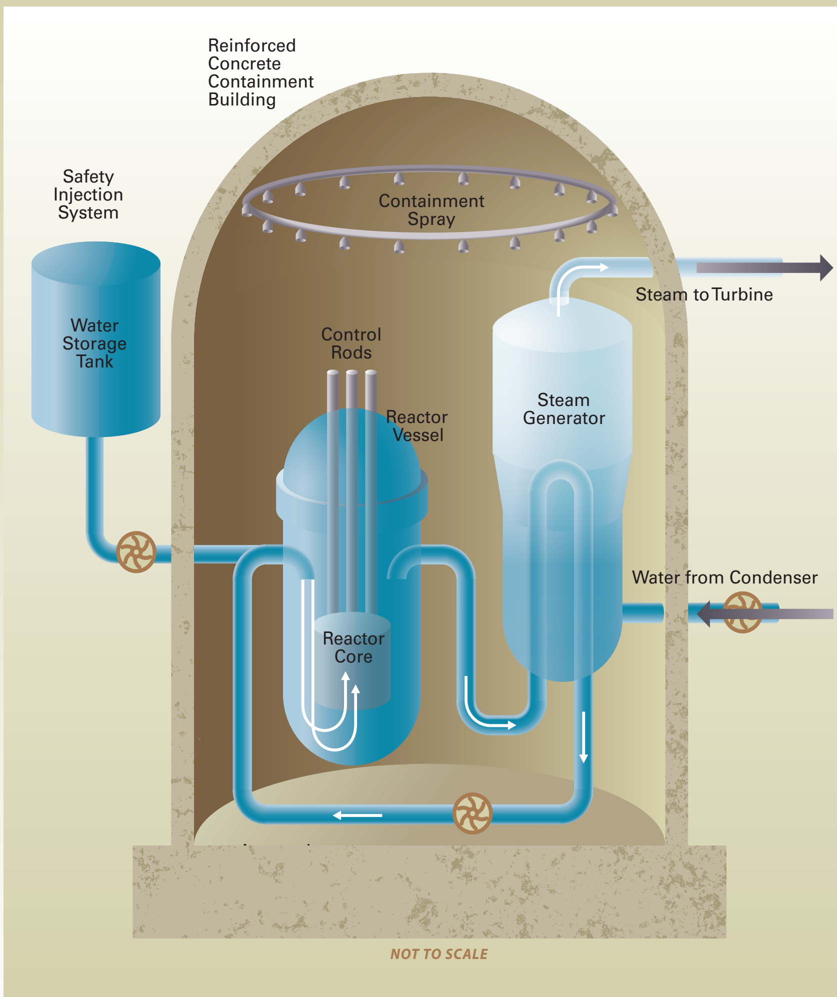
than 1 in a million reactor-years. SOARCA also sought to analyze scenarios leading to an early failure or bypass of the containment where the CDF is equal to or greater than 1 in 10 million reactor-years, since these scenarios have a potential for higher consequences and risk. This approach allowed a more detailed analysis of accident consequences for the more likely, although still remote, accident scenarios.

WHAT ACCIDENT SCENARIOS WERE ANALYZED?

For Peach Bottom, Surry and Sequoyah, the team modeled loss of all alternating current (ac) electrical power or "station blackout (SBO)" scenarios caused by earthquakes more severe than anticipated in the plant's design. SBO frequencies from flood or fire scenarios were combined with the earthquake frequency for scenario selection;

Figure 2.3 Typical U.S. Pressurized Water Reactor with a Large Dry Containment

A PWR has separate coolant loops to cool the reactor and generate the steam. The PWR's coolant loop (known as the primary loop) is under very high pressure (about 2,300 psi) to prevent water from boiling. The water is pumped through the reactor core where it is heated to about 600°F before being routed to the steam generators. The water travels through thousands of small tubes inside the steam generators where it heats secondary loop water at a lower pressure (about 900 psi) to produce steam at about 530°F. This steam enters the main steam line that routes it to the turbine generator. From the turbine generator, the steam then enters the condenser that cools it back to water so it can be pumped back to the steam generator to repeat the cycle. A typical PWR core has 150 to 250 fuel assemblies, and each assembly contains 200 to 300 fuel rods in a 14x14 to 17x17 matrix. Each PWR reactor has 2, 3, or 4 steam generators connected to it. The PWR in this figure is shown with a large dry containment. More information is available at http://www.nrc.gov/reactors/pwrs.html.



however, SOARCA modeled the earthquake scenario presents a severe challenge to the plant operators as well as offsite emergency responders, and has a high probability of occurring among the still unlikely severe accident scenarios possible.

Long-Term Station Blackout (LTSBO)—In this scenario, the plant is assumed to lose all ac power sources, but battery backups operate safety systems for about 4-8 hours until the batteries are exhausted.

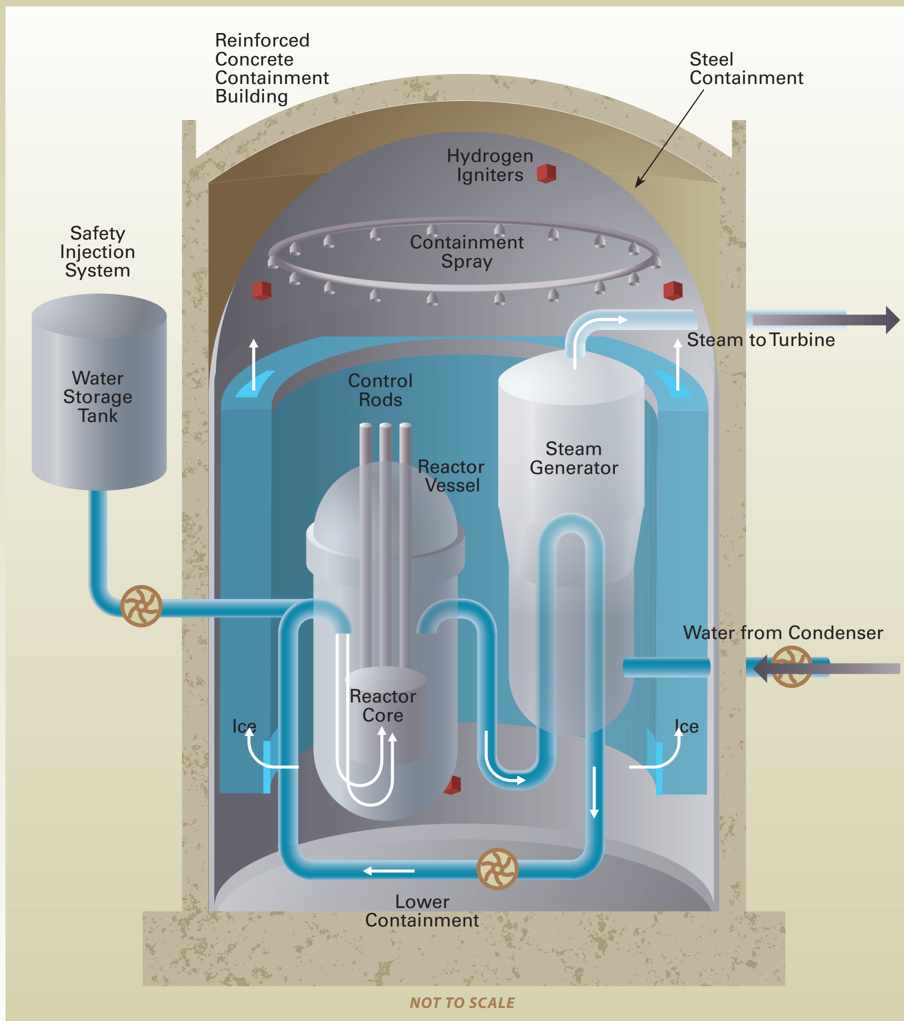
Short-Term Station Blackout (STSBO)—In this scenario, the plant is assumed to lose all power (even the batteries), all of the safety systems immediately become inoperable, and core damage occurs in the "short term."[1] In addition, the team analyzed two scenarios for Surry in which radioactive material could potentially reach the

What is a Station Blackout?

Reactor cooling systems at nuclear power plants are powered by alternating current (ac) power. This ac power is normally supplied by offsite power sources via the electrical grid but can be supplied by onsite sources such as emergency diesel generators if needed. A station blackout (SBO) involves the total loss of ac power when both offsite and onsite ac power sources fail. During an SBO, reactor cooling is temporarily provided by systems that do not rely on ac power, such as turbine-driven pumps that are driven by steam from the reactor. Batteries also are used to provide direct current (dc) power to control the turbine-driven pumps and to power instrumentation. Historically, risk models have indicated that the station blackout is an important contributor to overall nuclear power plant risk.

Figure 2.4 Typical U.S. Pressurized Water Reactor with an Ice Condenser Containment

Some PWRs have an ice condenser containment instead of a large dry containment. The ice condenser's smaller containment contains about 1,200 tons of ice around the perimeter of the containment. If steam and hot gases are released from the reactor vessel during a severe accident, they pass through the ice to the upper containment. The ice absorbs heat, condenses steam, and reduces pressure in the event of an accident. This helps the containment maintain its integrity. To prevent hydrogen from building up to dangerous levels in containment, igniters are installed in many locations to burn off hydrogen in small batches.



1 This terminology for long-term SBO and short-term SBO is consistent with that used in past NRC studies including NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants".

environment by bypassing containment features. These are described directly below and discussed in more detail in Chapter 4.

Interfacing Systems Loss-of-Coolant Accident (ISLOCA)—In this scenario, a random failure of valves ruptures low-pressure system piping outside containment that connects with the high-pressure reactor system inside containment.

Thermally Induced Steam Generator Tube Rupture (TISGTR)—This scenario is a lower probability variation of the STSBO. While the core is overheating and boiling off the available water, extremely hot steam and hydrogen circulating through the steam generator rupture a steam generator tube resulting in a pathway for radioactive material to escape to the non-radioactive portion of the plant and potentially to the environment.

Peach Bottom, Surry and Sequoyah each have two reactor units on the site. Multiunit accidents (events leading to reactor core damage at multiple units on the same site) could be caused by certain initiators such as an earthquake. Most PRAs and health consequence studies developed to date do not explicitly consider multiunit accidents because NRC policy is to apply the Commission's "Safety Goals for the Operation of Nuclear Power Plants" (51 FR 28044) and subsidiary risk acceptance guidelines on a "per reactor" basis. Therefore only single-reactor accidents were evaluated in SOARCA.

The NRC is currently performing a Level 3 PRA project that examines the risks associated with multi-reactor accidents at one site. This analysis is using many of the analysis tools used in SOARCA.

HOW WERE THE ACCIDENTS MODELED?

The SOARCA team modeled the accident scenarios and their potential to damage the core as realistically as possible by gathering detailed information about each of the three plants studied. The team asked plant staff for specific information about the design and operation of each plant system. The models' realism is enhanced by incorporating recent U.S. and international research about severe accidents and accounting for additional structures within containment (such as internal walls, piping, pumps, and heat exchangers) and buildings adjacent to the containment.

The state-of-the-art MELCOR computer code modeled how each scenario would unfold at each plant. The MELCOR results describe the following:

- How the plant and its emergency systems perform in response to an accident.
- How the reactor core behaves as it heats up beyond normal temperature limits.
- How the fuel itself, the reactor piping, and the containment building behave under high temperatures and pressures
- Whether radioactive material reaches the environment and, if so, how it occurs and how much material is released.

This information is based on the plant's design and physical safety systems. In addition, nuclear plants have a series of redundant and diverse safety measures to back up the designed safety systems. Chapter 3 of this brochure discusses how the SOARCA project models the actions that can potentially prevent or mitigate the release of radioactive material and ultimately protect the public. If a scenario caused a release of radioactive material, the team used another computer code (MACCS) to calculate the offsite health consequences of the release; Chapters 4, 5, and 6 provide more details about this step.

HOW LIKELY ARE THESE ACCIDENTS?

Overall, the SOARCA scenarios have core damage likelihoods that range from about 1 accident in 50,000 years to 1 accident in 30 million years of reactor operation. Table 2.1 shows the likelihoods for each scenario in order of more likely scenarios to less likely scenarios. Although the chances of these scenarios ever occurring are very small, probabilistic risk assessments have shown that these scenarios are very important core damage sequences.

SOARCA examines the effectiveness of actions to mitigate each accident (should one occur) and to prevent radioactive material from reaching the public and the environment. The likelihoods of the scenarios selected for SOARCA were based on: a review of NUREG-1150; the Individual Plant Examination of External Events (IPEEEs) conducted by licensees in the 1990s; NRC-developed SPAR models of external events; licensee-sponsored PRAs; and other NRC-sponsored studies. There was no attempt to match the stated likelihoods to any one particular study. Rather, they reflect the expert opinion of the NRC staff, based on all these sources of information available in 2005 for Peach Bottom and Surry, and 2014 for Sequoyah, when the scenarios were selected. Updated information could affect these estimates. For example, NRC staff expects to gain further insight into seismic and flooding event scenarios when U.S. nuclear power plants implement recommendations from the Fukushima Near-Term Task Force report (July 2011).

Table 2.1 Likelihoods of SOARCA Accident Scenarios

Reactor Site	Accident Scenario	Approximate Likelihood (per years of reactor operation)
Surry	Long-Term Station Blackout	1 event in ~ 50,000 years of reactor operation
Sequoyah	Long-Term Station Blackout	1 event in ~ 100,000 years
Peach Bottom	Long-Term Station Blackout	1 event in ~ 300,000 years
Surry, Sequoyah	Short-Term Station Blackout	1 event in ~ 500,000 years
Surry	Short-Term Station Blackout with Thermally Induced Steam Generator Tube Rupture	1 event in ~ 3 million years
Peach Bottom	Short-Term Station Blackout	1 event in ~ 3 million years
Surry	Interfacing Systems Loss-of-Coolant Accident	1 event in ~ 30 million years

Historical Perspective: Three Mile Island and Chernobyl

Many people are familiar with the Three Mile Island (pictured left) and Chernobyl (pictured right) accidents. Although SOARCA did not examine these historical accidents, this brochure provides information about them so readers can compare the results of this research study to real events.

On March 28, 1979, the Three Mile Island accident occurred in Pennsylvania as a result of equipment malfunctions, design-related problems, and worker errors. The accident melted almost half the reactor core of Unit 2 and released contaminated water, hydrogen gas, and radioactive material into the containment building. A very small amount of radioactive material reached the environment. TMI had hydrogen burns, also modeled in SOARCA, but in the TMI accident the contaiment did not fail. It remains the most serious accident in U.S. commercial nuclear power plant operating history although no plant workers or members of the nearby community were injured or killed. A long-term follow-up study by the University of Pittsburgh that evaluated local, county, and State population data from 1979 through 1998 concluded that there is not an increase in overall cancer deaths among the people living within a 5-mile radius of Three Mile Island at the time of the accident. This accident brought about sweeping changes for nuclear power plants and heightened oversight by NRC.

On April 26, 1986, an accident destroyed Unit 4 of the nuclear power station at Chernobyl, Ukraine, in the former USSR. The series of events that led to this accident could not occur at U.S. commercial power reactors because U.S. reactors have different plant designs, robust containment structures, and operational controls to protect them against the combination of lapses that led to the accident at Chernobyl. Its operators ran an experiment that led to a sudden surge of power, destroying the reactor core and releasing massive amounts of radioactive material into the environment. About 30 emergency responders died in the first 4 months after the accident. The health of the evacuated population and populations in contaminated areas of Belarus, the Russian Federation, and Ukraine has been monitored since 1986. Monitoring efforts to date indicate that a lack of prompt countermeasures resulted in increased risk of thyroid cancer to members of the public, most notably among people who were children or young adults at the time of the accident. No other health effects are attributed to the radiological exposure in the general population. Chernobyl's design, which differed significantly from reactors operating in the United States, made it vulnerable to such a severe accident.

NRC Fact Sheets about Three Mile Island and Chernobyl Accidents are available at:

- http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/3mile-isle.html
- http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/chernobyl-bg.html.

Historical Perspective: Fukushima Dai-ichi and NRC Response

On March 11, 2011, a 9.0-magnitude earthquake struck Japan about 231 miles northeast of Tokyo off the east coast of Honshu Island. The earthquake led to the automatic shutdown of 11 reactors at 4 sites (Onagawa, Fukushima Dai-ichi, Fukushima Dai-ni, and Tokai). At Fukushima Dai-ichi, which includes General Electric BWR Mark I reactors similar to the Peach Bottom plants, diesel generators provided electricity to plant systems until about 40 minutes later. At that point, a tsunami, estimated to have exceeded 45 feet (14 meters) in height, appeared to have caused the loss of all alternating current (ac) power and most emergency diesel generators to the six Fukushima Dai-ichi reactors. Three Fukushima Dai-ichi reactors (Units 1-3) were in operation at the time of the earthquake, and three (Units 4-6) were shut down for routine refueling and maintenance. Loss of ac power to pump water into Units 1 through 3 to cool the nuclear fuel resulted in melted fuel and severely damaged cores. The melted fuel cladding reacted with steam and generated hydrogen gas. The hydrogen reached critical levels and caused explosions. The reactor damage, along with hydrogen gas explosions inside the units, released radioactive material into the environment. The earthquake and tsunami devastation in the area significantly delayed offsite assistance. Additional systems were finally able to use seawater to cool the reactors.

Since the events at Fukushima began to unfold, NRC has been working to understand the events in Japan and relay important information to U.S. nuclear power plants. Not long after the emergency began, NRC established a task force of senior NRC experts to determine lessons learned from the accident and to initiate a review of NRC regulations to determine if additional measures should be taken immediately to ensure the safety of U.S. nuclear power plants. The task force issued its report on July 12, 2011, concluding that continued U.S. plant operation and NRC licensing activities presented no imminent risk. The Task Force also concluded that enhancements to safety and emergency preparedness are warranted and made a dozen general recommendations for Commission consideration. The NRC is currently implementing many of those recommendations to enhance U.S. nuclear plant safety. The NRC issued a Mitigation Strategies Order, EA-12-049, on March 12, 2012, requiring all U.S. nuclear power plants to implement strategies that will allow them to cope without their permanent electrical power sources for an indefinite amount of time. The licensees are implementing the strategies and making plant modifications.

An appendix to the main SOARCA report (NUREG-1935) briefly compares and contrasts what we currently know about Fukushima with insights from the Peach Bottom SOARCA analyses. Because the Fukushima accident was in some ways similar to a few of the Peach Bottom scenarios analyzed in SOARCA, the SOARCA Peach Bottom MELCOR model was used by the NRC and the U.S. Department of Energy (DOE) to help evaluate the Fukushima accident as it was unfolding (see box directly below.) The NRC Web site has additional information on the Fukushima accident and NRC's response:

- https://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard.html

Comparison of Fukushima Accident to SOARCA Analyses

Following the Fukushima accident of 2011, the U.S. Department of Energy (DOE) and the NRC began a cooperative effort to use the MELCOR code for a forensic analysis of event progression to develop a more detailed understanding of the accident. This cooperative effort is ongoing. The SOARCA team compared and contrasted the Fukushima accident and the SOARCA study for the following topics: (1) operation of the reactor core isolation cooling (RCIC) system, (2) hydrogen release and combustion, (3) 48-hour truncation of releases in SOARCA, (4) multiunit risk, and (5) spent fuel pool risk. It must be emphasized that we need much more information to be certain about what actually occurred in the Fukushima reactors. Our current uncertainty prevents us from drawing firm conclusions regarding comparisons with SOARCA results.

As the NRC learned more about the damage to plant safety functions gathered over the weeks and months following these events, many similarities became apparent between SOARCA's calculated damage progression in the Peach Bottom SBO accident scenarios and the progression of events at Fukushima. These similarities include the following:

- the sequence and timing of events that followed the loss of core cooling, including the start of core damage and radioactive material release from the fuel,
- challenges to containment integrity from the loss of fuel heat removal and the accumulation of hydrogen generated during fuel damage within the reactor vessel, and
- the destructive effects of hydrogen combustion in the reactor building.

Some notable differences were also obvious between the events that unfolded at Fukushima and the Peach Bottom LTSBO scenario studied in the SOARCA project. These differences, for example the use and timing of certain safety systems, led the NRC staff to take a closer look at the models used and assumptions made in the LTSBO analyses. SOARCA analysis results were qualitatively compared to the preliminary events and information available in the evaluation of the Fukushima Dai-ichi accident. SOARCA's conclusions remain valid in light of information currently available from the events that unfolded at Fukushima.

ACTIONS TO MITIGATE ACCIDENTS

CHAPTER 3

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

Defense-in-Depth Philosophy

"Defense in depth" is NRC's approach to approve the design and operation of nuclear facilities to prevent and mitigate accidents that could release radioactive materials. The key is creating multiple independent and redundant layers of defense to compensate for potential human and mechanical failures so that no single layer, no matter how robust, is exclusively relied upon. Defense-in-depth includes the use of redundant and diverse key safety functions and emergency response measures. For further information, see Speech No. S-04-009, "The Very Best-Laid Plans (the NRC's Defense-in Depth Philosophy)."

HOW CAN POTENTIAL ACCIDENTS BE MITIGATED?

In addition to the redundant and diverse physical systems designed to prevent accidents, NRC and plant owners understand the importance of having preplanned emergency measures in the unlikely event an accident occurs. NRC expects these emergency measures will mitigate accident consequences by preventing core damage or preventing, delaying, or reducing the release of radioactive material. NRC requires plant operators to maintain detailed emergency procedure plans for the entire range of possible accidents. These plans include the following:

Emergency operating procedures—These procedures list operator actions to mitigate possible nuclear power plant emergencies.

Severe accident management guidelines—These are operator guidelines to mitigate accidents that are more severe than what the facility was designed to handle.

10 CFR 50.54(hh)(2) mitigation measures—These measures include plans and resources that nuclear plants put in place to meet additional NRC requirements following the terrorist attacks on September 11, 2001 to mitigate scenarios involving the loss of large areas of the plant due to explosions or fire. The initiating event may include other natural phenomena like earthquakes.

WHAT ASSURANCE DOES NRC HAVE THAT THESE MITIGATING ACTIONS CAN WORK?

NRC requires its licensees to train and practice emergency operating procedures in simulators that replicate the plant control rooms at each site. NRC also requires that plant owners have developed severe accident management guidelines and implemented the security-related mitigation measures to ensure that they have proper equipment, procedures, and training. NRC inspectors observe these activities to ensure NRC regulations are met at each plant.

HOW ARE MITIGATING ACTIONS MODELED?

SOARCA is the first detailed analysis that quantifies the value of the 10 CFR 50.54(hh)(2) mitigating actions in responding to potential accident conditions. This equipment and procedures for maintaining or restoring safety functions after explosions or fire, have been further enhanced by NRC actions after Fukushima. The NRC concludes plant operators could use this equipment for other types of accidents.

Therefore, 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 are fully successful in carrying out the mitigating actions. The project team accomplished this by holding tabletop exercises with senior reactor operators and emergency response personnel at Peach Bottom, Surry and Sequoyah to determine what actions would be taken to mitigate each scenario analyzed including the time required to implement each action. Many of these actions are designed to help in the case of large fires and explosions but could potentially be used for the scenarios analyzed in SOARCA.

The Surry mitigated LTSBO showed that the accident could be mitigated prior to core damage and containment failure. Given the high degree of similarity between Surry and Sequoyah within the reactor core and coolant system, the same type of mitigated case for the Sequoyah LTSBO, in which operators are modeled as fully successful in carrying out mitigating actions, would not have shown any new insights. Therefore the mitigated case assumption for Sequoyah was different in that only hydrogen igniters are credited following reactor core damage.

Unmitigated Case—To understand the value of 10 CFR 50.54(hh)(2) mitigating actions and to provide a basis for comparing SOARCA results to past studies, the team also analyzed an "unmitigated case" for each scenario. These unmitigated cases assumed that the plant failed to implement 10 CFR 50.54(hh)(2) measures. Although the earthquakes considered in the SOARCA scenarios exceed the plants' designs, the more rugged engineered safety features are assumed to survive in both the unmitigated and mitigated cases. The unmitigated cases modeled the sequence of events where operator actions in using these safety features are unsuccessful and lead to fuel damage, release of radioactive materials, and offsite health consequences.

WHAT IS THE TIMING OF MITIGATING ACTIONS?

Detailed MELCOR modeling demonstrated that plant operators can have time during accident scenarios to perform the necessary emergency actions. Figures 3.1 through 3.3 compare the mitigated and unmitigated timelines for the Peach Bottom and Surry long-term station blackout scenarios until the release starts (for the unmitigated case).

Historical Perspective: Examples of Improvements in Mitigation Capabilities Since 9/11

In response to the terrorist attacks of September 11, 2001, NRC and operating reactor licensees worked together to develop improved mitigation methods for events that could disable large areas of a nuclear power plant. As a result, operating reactor licensees purchased equipment and developed procedures for each site to better mitigate such events. NRC codified the requirements for this additional mitigation in Title 10 CFR 50.54 (hh)(2). These mitigation measures include the following for the Peach Bottom and Surry plants:

- Portable diesel-fuel powered pumps (pictured).
- Portable generators to provide electricity to power critical instrumentation and to open or close valves.
- Prestaged air bottles to open or close air-operated valves.
- Procedures for operating steam-turbine-driven pumps without power.
- Designated make-up water sources.

PRAs commonly include a human reliability analysis to represent the likelihood of operator actions. SOARCA evaluated human actions through tabletop exercises, walkdowns, simulator runs, and other inputs from licensee staff.

Figure 3.1 Comparison of SOARCA Accident Progression Timing for Mitigated and Unmitigated Cases of Peach Bottom Long-Term Station Blackout [NUREG/CR-7110, Volume 1, Table 5-1]

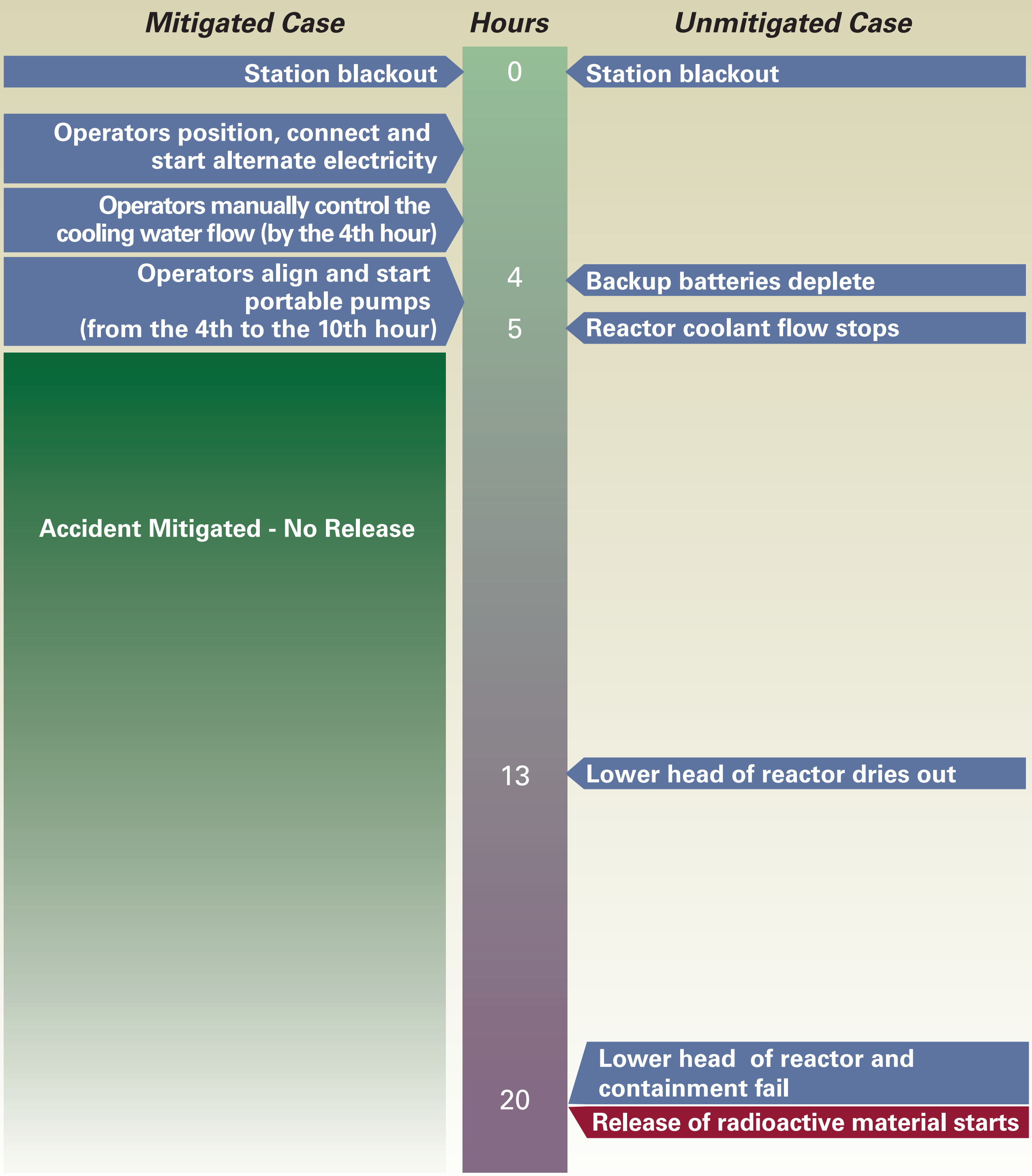


Figure 3.2 Comparison of SOARCA Accident Progression Timing for Mitigated and Unmitigated Cases of Surry Long-Term Station Blackout [NUREG/CR-7110, Volume 2, Table 5-1]

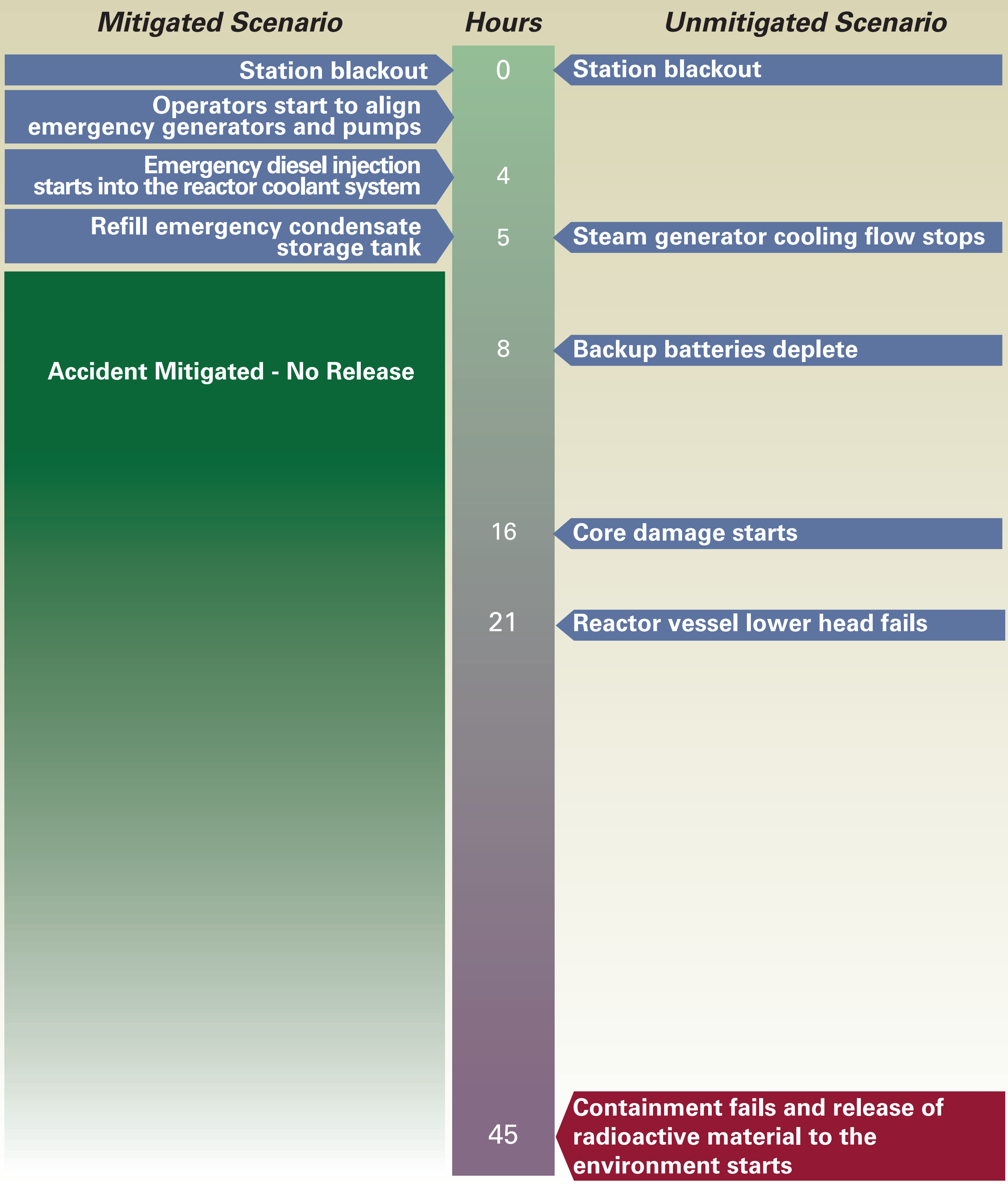
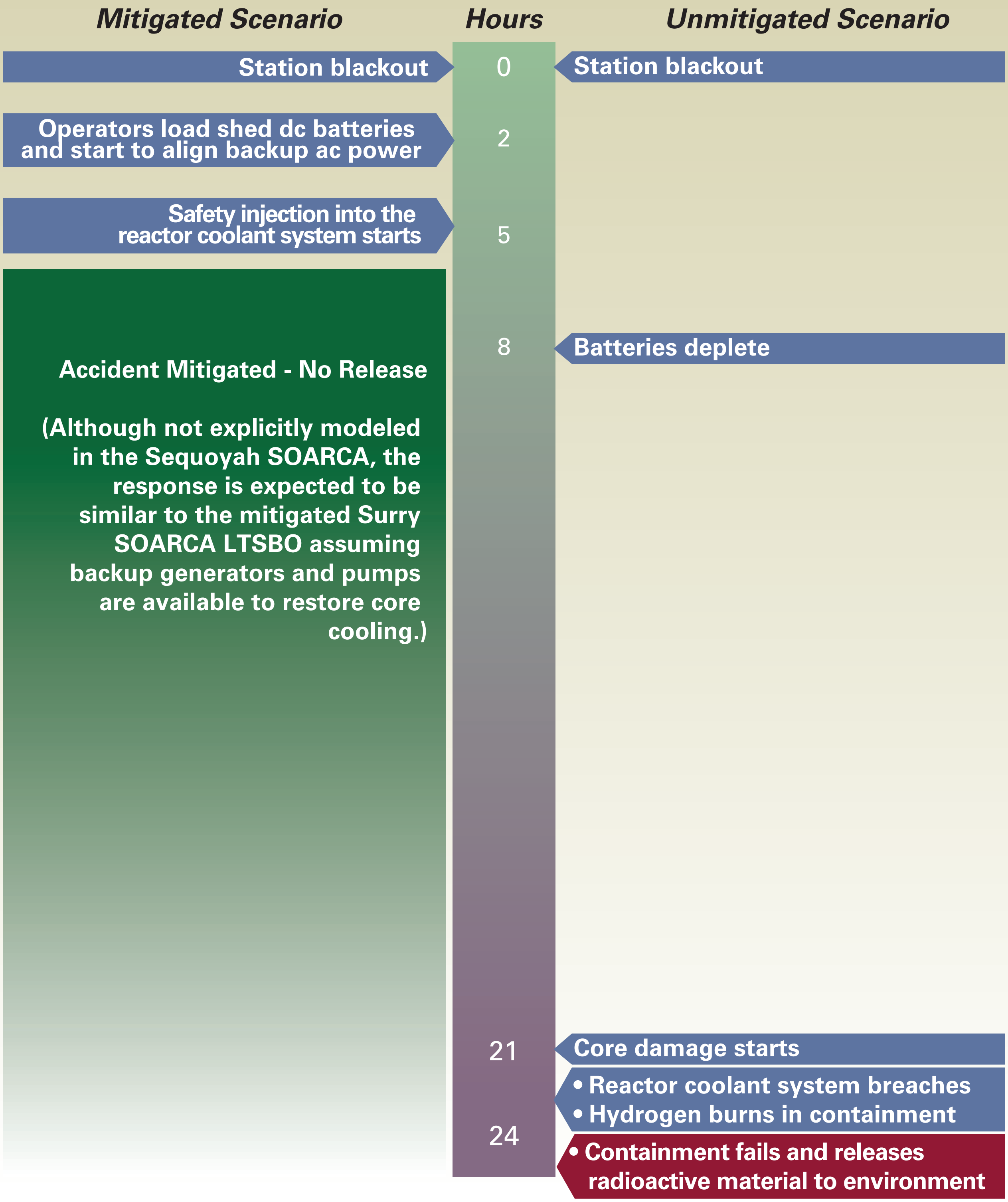


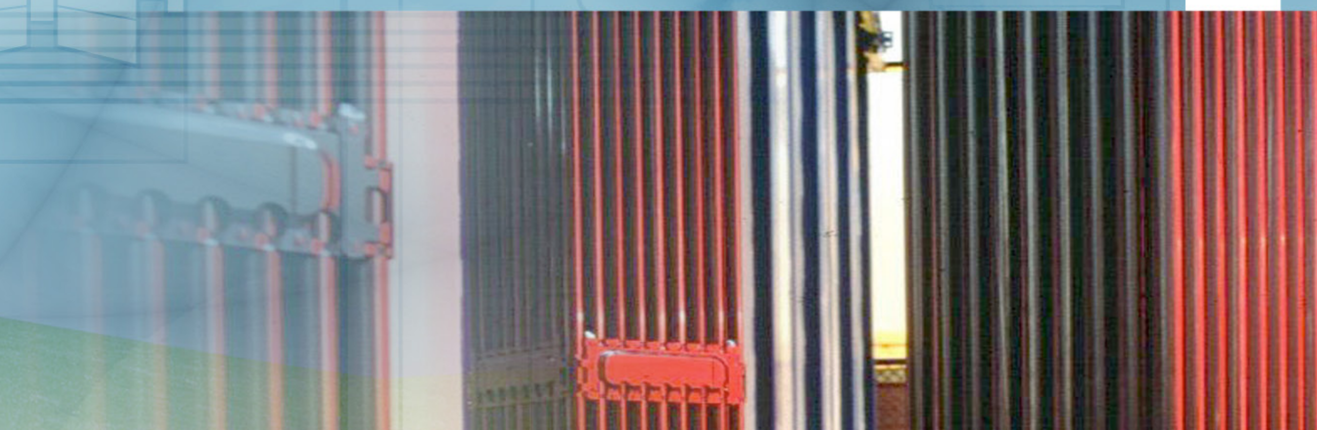
Figure 3.3 Comparison of SOARCA Accident Progression Timing for Mitigated and Unmitigated Cases of Sequoyah Long-Term Station Blackout [NUREG/CR-7245, Figure 4-163]



These results are sensitive to the rate of hydrogen release, presence of ignition sources, and steam and oxygen concentrations. For LTSBO, containment failure is possible after 24 hours due to hydrogen burn.

RELEASE OF RADIOACTIVE MATERIAL

CHAPTER 4



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

The SOARCA models showed that mitigating actions can prevent core damage or reduce or delay a release of radioactive material. For the scenarios examined, the SOARCA team also modeled unmitigated cases that lead ultimately to a release to the environment. The MELCOR computer code models the behavior of radioactive materials to the point that they escape from containment.

WHAT RADIOACTIVE MATERIAL DOES SOARCA MODEL?

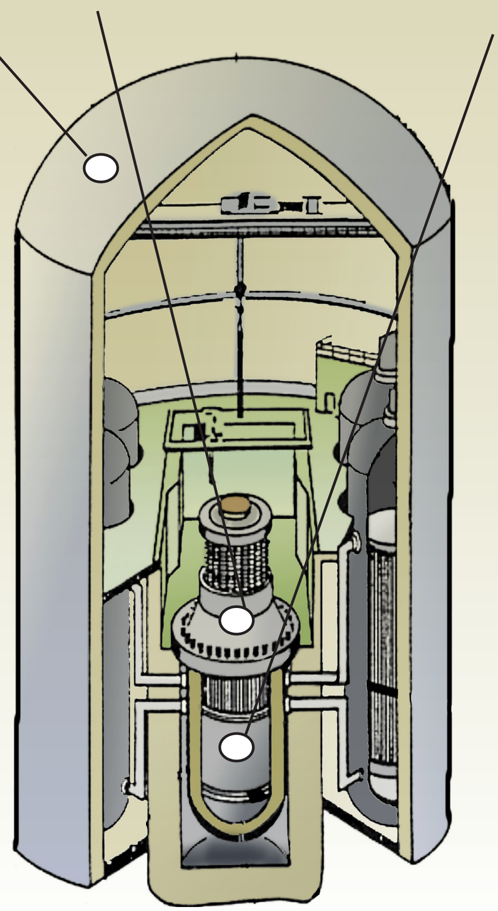
SOARCA took a detailed approach to considering radioactive substances, or radionuclides. In SOARCA, MELCOR calculations of reactor accident response are based on realistic estimates of decay heat generated by the radionuclides in the reactor core. MELCOR organizes the radioactive material by chemical similarity to track them as they are released from the reactor core and move through piping, the containment building, and other buildings on their way to the environment. A significant part of the radioactive material may settle inside the containment building prior to containment failure and release into the environment surrounding the plant site. The offsite consequences

How Does Containment Work?

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

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

Reactor Vessel— metal enclosure that holds the reactor core surbmerged in cooling water.



Fuel Rods—long, slender tubes that hold uranium fuel for nuclear reactor use. Fuel rods are assembled into bundles that are loaded individually into the reactor core (see image below).

Note: Typical large dry containment shown.

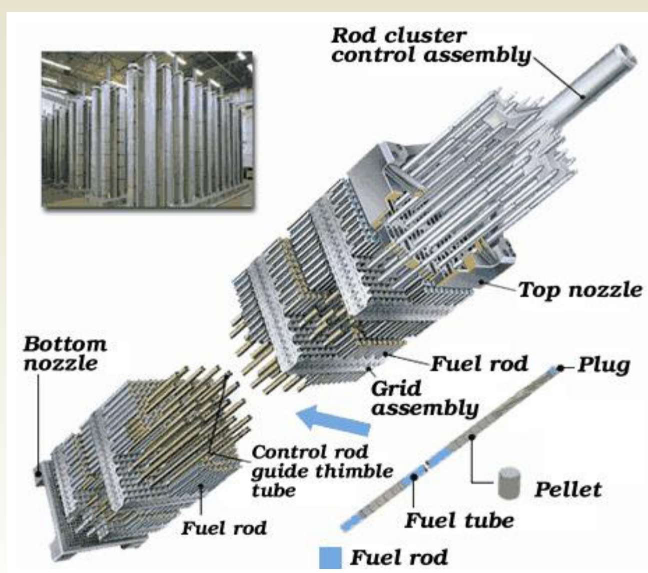


Diagram of components of a reactor fuel assembly computer code (MACCS) tracks radionuclides based on how long they remain radioactive, their biological importance, and how much is expected to be released from the core.

Cesium and Iodine—These two radioactive material groups affect offsite consequence analysis because they are released as part of an accident, and the human body can get significant radiation doses from them.

Other Radioactive Material—MELCOR and MACCS also consider other radioactive material inventory in the analysis, and consequence results in NUREG-1935 include health effects from the radioactive material released in the accident.

WHAT INFORMATION IS INCLUDED IN MELCOR MODELING?

In MELCOR modeling we look at:

- How physical and chemical processes influence the behavior of radioactive material while the core heats up.
- How the accident's extremely high temperatures influence particles' behavior at the molecular level and their physical states (e.g., turning them into gas or small particles that can settle or move through the air).
- How the radioactive material moves within the containment and reactor coolant system (before exiting containment).
- How engineered safety systems (such as water sprays and air fan coolers) impact the behavior of radioactive material to prevent their release.
- If and at what rate the accident releases radioactive material into the environment.

HOW ARE RADIOACTIVE MATERIALS MODELED TO ESCAPE FROM CONTAINMENT?

The following sections describe the timing of radioactive material movement while onsite and its release to the environment. Figure 4.1 shows how much of the reactor core's available radioactive iodine (I-131) and cesium (Cs-137) is released to the environment.

Peach Bottom Scenarios (Unmitigated Cases)

Peach Bottom's containment is comprised of a concrete enclosure with a steel liner, which will eventually heat up and pressurize during an unmitigated severe accident. This can result in leakage of radioactive material to the environment. The heat and pressure will eventually result in melting the steel liner shell and cracking the concrete enclosure.

Long-Term Station Blackout— About 19 hours after the scenario begins, molten core material penetrates 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. An uncertainty analysis of this scenario was run showing a range of time to containment failure of 12-24 hours. [NUREG/CR-7155, Figure ES-1, ES-2 median value and 5th to 95th percentile range]

Short-Term Station Blackout— About 8 hours after the scenario begins, molten core material penetrates 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. [NUREG/CR-7110, V1 Table 5-4]

For the two analyzed Peach Bottom station blackout events, while the core is in the reactor vessel, radioactive material moves from the core into the bottom of the suppression pool as relief valves send steam into the suppression pool. Some material deposits on reactor vessel and pipe surfaces on its way to the suppression pool; the rest is retained in the suppression pool as the steam is condensed in the pool.

Surry Scenarios (Unmitigated Cases)

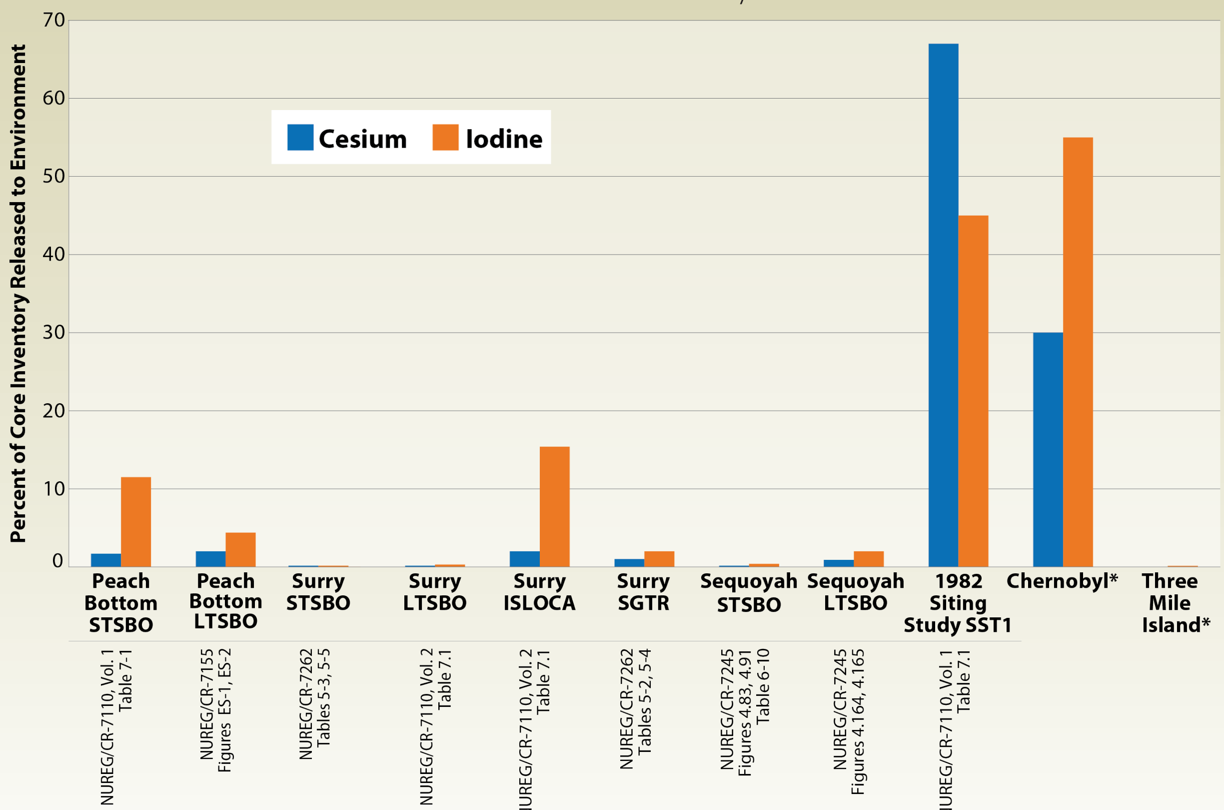
Surry has a reinforced concrete containment with a steel liner that can eventually heat up and pressurize, resulting in leakage and ultimately failure of the containment building, melting the liner and cracking the concrete.

Long-Term Station Blackout— About 45 hours after the scenario begins, the pressure in the containment building exceeds the building's limits, tearing the containment liner and cracking the reinforced concrete.[NUREG/CR-7110, V2, Table 5-1]

Short-Term Station Blackout— An uncertainty analysis of this scenario showed that about 50 hours (median value) after the scenario begins, the pressure in the containment exceeds the building's limits, tearing the containment liner and cracking the reinforced concrete. The uncertainty analysis showed a range of time to containment failure of 34 – 72+ hours. Radionuclide release to the environment can

Figure 4.1 Percentages of Cesium and Iodine Released to the Environment for SOARCA Unmitigated Scenarios, 1982 Siting Study (SST1), and Historical Accidents

The SOARCA unmitigated release of Cesium-137 and Iodine-131, for each of the modelled scenarios, are much smaller than estimated in the earlier 1982 Siting Study Source Term 1 (SST1) case. Some of these releases develop over a period of time and are also much smaller than those from the Chernobyl accident.



* Chernobyl release data is estimated at 20-40 percent for cesium-137 and 50-60 percent for iodine-131. Three Mile Island released an extremely small quantity of iodine-131 (~ 15 curies) and zero cesium-137. Fukushima releases are estimated to be approximately one-tenth of releases from Chernobyl [IAEA Report GC(59)/14].

start much earlier after core damage through the intact containment leakage pathway, however this is very small relative the release that occurs when the containment fails. [NUREG/CR-7262 page xxi and Table 5-13].

Interfacing Systems Loss-of-Coolant Accident—The scenario begins with the hypothesized random failure of 2 valves in series, rupturing a pipe outside of the containment building. This provides a path from the reactor core to the environment which bypasses containment. About 13 hours after the scenario begins, the accident progresses to the point where the fuel overheats and gaseous radioactive particles are released through this path. When the overheating fuel is in the reactor vessel, some of the radioactive material moves from the fuel through the ruptured pipe and into the safeguards building. Most of this radioactive material deposits on reactor vessel and pipe surfaces and safeguards building (next to containment) filters, with a fraction of it entering the environment. [NUREG/CR-7110, V2 Table 5-14].

Short-Term Station Blackout with Thermally Induced Steam Generator Tube Rupture— About 3.5 hours after the scenario begins, high-pressure, high-temperature gas circulating through the reactor coolant system ruptures a steam generator tube, a steam generator safety relief valve is opened, allowing gaseous radioactive particles to flow out of the broken tube bypassing the containment building. This rupture creates about a 1-inch diameter hole. Minutes later, a reactor coolant system pipe also ruptures—creating about a 2-foot diameter hole. In the period of time between the two ruptures, much of the radioactive material deposits in the failed steam generator, and this settling helps prevent much of it from flowing out into the environment. After the pipe rupture, the radioactive material primarily flows into and deposits in the containment. An uncertainty analysis showed a range of steam geneator rupture times of 3.5 - 18 hours. [NUREG/CR-7262, Figure ES-1]

Sequoyah Scenarios (Unmitigated Cases)

Sequoyah has a steel containment that can heat up and eventually fail due to sudden pressure spikes caused by hydrogen burns or due to slow pressurization caused by molten core-concrete interaction.

Long-Term Station Blackout— A sensitivity analysis of this scenario shows two potential containment outcomes: failure at about 24 hours due to hydrogen burns inside the containment causing an increase in pressure that exceeds the containment's failure limits, tearing the steel containment wall, or late failure after about 72 hours due to slow but gradual overpressure. [NUREG/CR-7245, Figure 4-163].

Short-Term Station Blackout— An uncertainty analysis of this scenario shows two potential containment outcomes: early failure (about 4 - 12 hours from the initiating event) due to hydrogen combustion or late failure (after 40 hours) from more gradual overpressure. [NUREG/CR-7245, Figure ES-2].

For the analyzed Surry and Sequoyah station blackout events, while the fuel is overheating, radioactive material enters the containment building through ruptured reactor coolant system piping or reactor vessel bottom head. Some material deposits on the inside surfaces of the reactor coolant system as it moves to the containment building. The remaining contained material deposits in the containment building.

Table 4.1 Timing and Quantity of Radioactive Material Released for SOARCA Mitigated and Unmitigated Scenarios

	From the initiating event, about how long until radioactive material is released to the environment?		About how much of the available radioactive material (Iodine-131 and Cesium-137) is released?	
	Mitigated Case	Unmitigated	Mitigated Case	Unmitigated
Peach Bottom Long-Term Station Blackout[a]	no release	19 hours (12 - 24 hours)	no release	Iodine: 4% (2%–13%) Cesium: 2% (1%–9%)
Peach Bottom Short-Term Station Blackout	no release	8 hours	no release	Iodine: 12% Cesium: 2%
Surry Long-Term Station Blackout	no release	45 hours	no release	Iodine: 0.3% Cesium: <0.01%
Surry Short-Term Station Blackout	no release modeled in MACCS	50 hours (34 - 72 hours)	no release within 48 hours[c]	Iodine: 0.03% (<0.01%–0.14%) Cesium: <0.01% (<0.01%–0.03%)
Surry Steam Generator Tube Rupture	3.5 hours	3.5 hours (3.5-18 hours)	Iodine: <1% Cesium: <1%	Iodine: 2% (0.6%–4%) Cesium: 1% (0.4%–2%)
Surry Interfacing Systems Loss-of-Coolant Accident	no release	13 hours	no release	Iodine: 16% Cesium: 2%
Sequoyah Long-Term Station Blackout	no release expected[d]	see note[b]	no release expected[d]	Iodine: 2% Cesium: 1%
Sequoyah Short-Term Station Blackout	no release expected[d]	58 hours (4-72+ hours)[b]	no release expected[d]	Iodine: 0.4% (0.4%–11%) Cesium: 0.1% (0.1%–4%)

[a] Ranges of values for the 5th - 95th percentile range are shown in parentheses where they are available from the uncertainty analyses (discussed further in Chapter 7), which were completed for some of the scenarios.

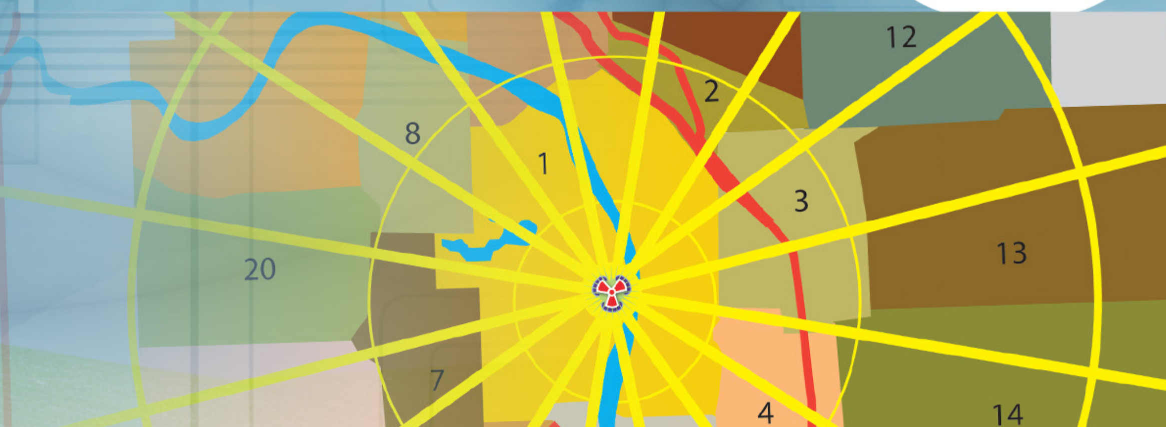
[b] For the Sequoyah unmitigated Short-Term Station Blackout, there are two potential containment outcomes: early failure (about 4-12 hours from initiating event) due to hydrogen combustion or late failure (greater than 40 hours) from more gradual overpressure due to molten core concrete interactions (CCI). For the unmitigated Long-Term Station Blackout, there is a chance of containment failure at about 24 hours due to a hydrogen burn, otherwise the containment failure can be delayed for as long as 72 hours or beyond due to slow pressurization.

[c] For the mitigated Surry STSBO, the reactor vessel would fail; however, the containment would not fail until about 66 hours after the blackout. A review of available resources and emergency plans shows that adequate mitigation measures could be brought onsite within 24 hours and connected and functioning within another 24 hours. Therefore, 66 hours would allow time for mitigation through measures transported from offsite, and this mitigation would avert containment failure such that radioactive material would not be released to the environment.

[d] Based on analogous Surry analysis, as explained above.

MODELING EMERGENCY PLANS

CHAPTER 5



This chapter explains emergency planning and how emergency response was modeled.

For scenarios leading to core damage and subsequent release of radioactive materials to the environment, the local public may be evacuated and/or sheltered. SOARCA models tracked the dispersion of radioactive material and analyzed the effect of carrying out emergency response for these scenarios. This chapter provides more information about how the SOARCA project modeled emergency plans during a severe accident.

WHAT IS EMERGENCY PLANNING?

NRC requires nuclear power plants to have onsite and offsite emergency plans as a defense-in-depth measure. NRC evaluates the plants' emergency planning to ensure they can execute their plans and coordinate State and Federal responses. Emergency plans focus on protecting public health and safety with the following objectives:

Onsite Objective—Stop the accident. NRC requires the utilities to have onsite response that includes technical, maintenance, and management staff that can respond within an hour of the accident's start. Each year, the licensees train and drill this capability, and NRC inspects it.

Offsite Objective—Protect the local population through implementation of protective actions that include evacuating and sheltering. NRC requires utilities to have offsite response support from local and State agencies. The Federal Emergency Management Agency inspects this capability every 2 years. Emergency planning zones (EPZs) help define where detailed protective strategies would be used during an emergency. Every plant must have NRC-approved emergency action levels that dictate declaring an emergency well before a severe accident could cause a core melt or radiation release. This timing is designed to ensure that emergency plans are implemented before the plant is in a serious state and that members of the public are well on their way to evacuation before any release begins.

WHAT INFORMATION IS INCLUDED IN EMERGENCY PLAN MODELING?

The SOARCA team modeled the specific emergency plans for each site using detailed information that included the following:

- Population based on data from the most recent U.S. Census available, and projected to the year of the analysis[1].
- Evacuation time estimates from emergency plans.
- Plans to relocate populations from contaminated areas.

Using each site's emergency plan information, the SOARCA team organized the population into several groups and modeled each group's evacuation timing along with the timing of the accident. Table 5.1 provides a description of the groups.

[1] *2000 US Census data projected to 2005 was used for Surry and Peach Bottom and 2010 US Census data projected to 2015 was used for Sequoyah.*

What Are NRC Regulations?

Emergency Plans

The planning standards of 10 CFR 50.47, "Emergency Plans," require nuclear plant licensees to develop comprehensive emergency response plans that include the support of State and local response organizations. Licensees must establish procedures to immediately notify offsite authorities of an emergency and establish warning systems to provide early notification and clear instruction to the public. Licensees must demonstrate to NRC that protective measures can and will be implemented in the event of a radiological emergency. For details, see http://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-0047.html.

Table 5.1 Evacuation Groups	
Schools	School populations within 10 miles of the site
General Public	People within 10 miles of the site who evacuate in response to the evacuation order
Special Facilities	Special-needs population, including residents of hospitals, nursing homes, assisted living communities, and prisons within 10 miles of the site
Nonevacuating Public	A portion of the public within 10 miles of the site who refuse to evacuate (assumed to be 0.5 percent of the population)
Shadow	Shadow evacuation occurs when members of the public evacuate from areas that are not under official evacuation orders, typically beginning when a large- scale evacuation is ordered
Tail	The last 10 percent of the public to evacuate from the 10-mile EPZ

WHAT DOES MODELING DEMONSTRATE ABOUT EMERGENCY PLANNING?

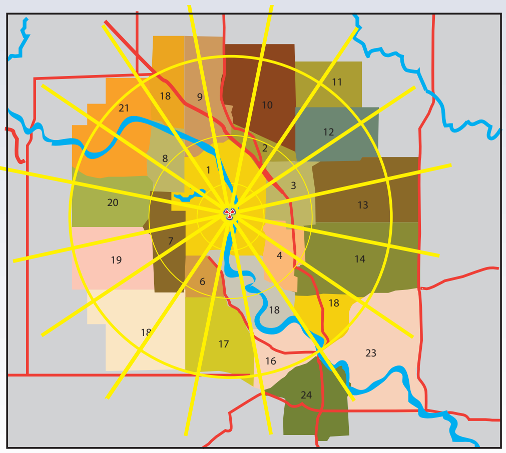
The MACCS computer code calculates the radiation dose to the public based on evacuating, sheltering, and returning to the area after the event. Figure 5.1 is an illustrative example of the modeled timing of the unmitigated Peach Bottom LTSBO base case scenario and the timing of emergency response. Because this analyzed accident scenario takes several hours to start releasing radioactive material to the environment, it provides time for the population to evacuate before potential radiation exposure. The analysis considered seismic impacts on emergency response (e.g., loss of bridges, traffic signals, and delayed notification). However, the MACCS modeling showed that seismic impacts for the sites did not significantly impact risk calculations because seismic impacts only affect the immediate phase of the accident when people are likely sheltering or evacuating. SOARCA's risk calculations are dominated instead by long- term exposure of the population after they return home when told it is safe to do so.

What Are Emergency Planning Zones (EPZs)?

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 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. The detailed planning for the EPZs enables emergency responders to extend actions beyond the EPZ if conditions warrant.

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 can include sheltering, evacuating, and taking potassium iodide pills to protect people who inhale or ingest airborne radioactive iodine.

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 can include a ban of contaminated food and water to protect people from radioactive material in the food chain. Ingestion of contaminated food and water is not treated in the SOARCA analyses because adequate supplies of food and water are available in the United States and can be distributed to areas affected by a reactor accident.

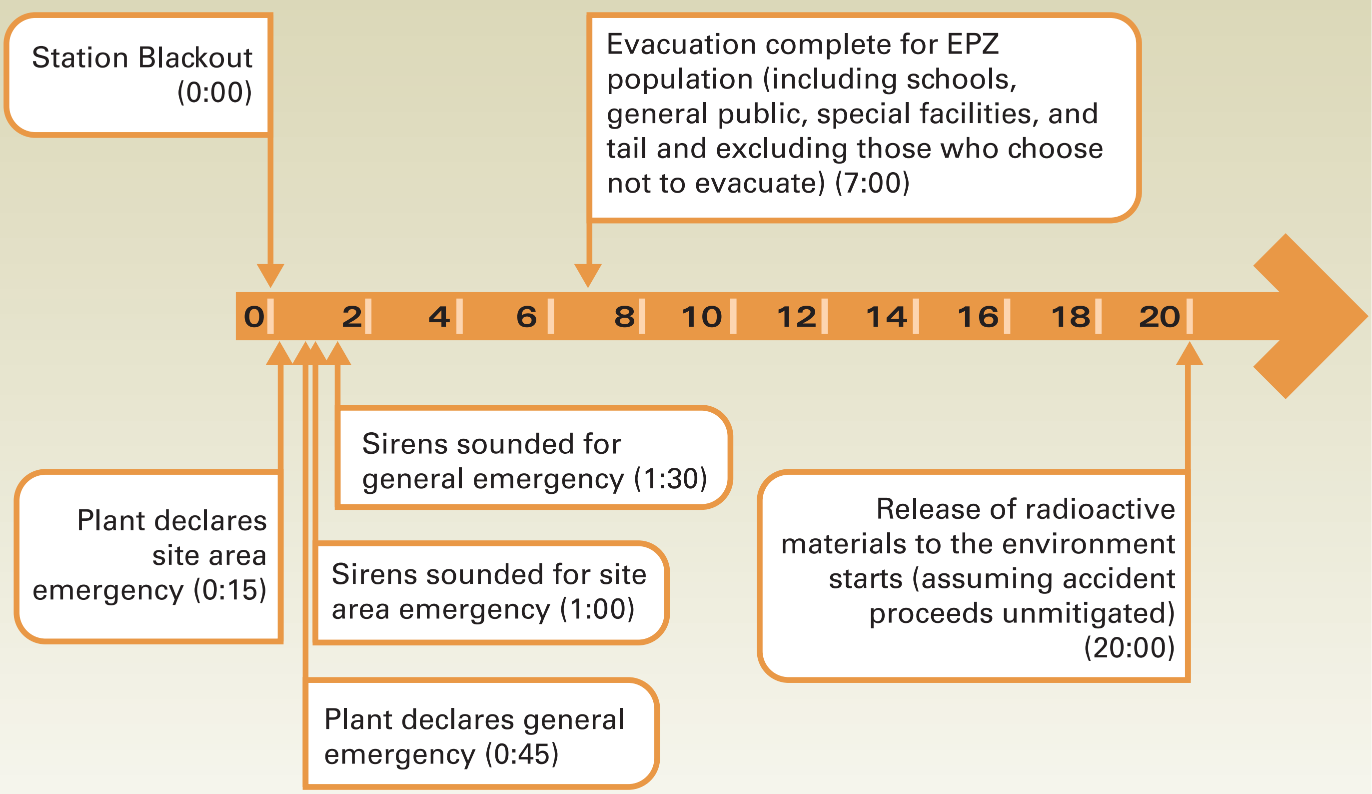


NRC Staff during an emergency preparedness drill

Figure 5.1 shows that groups are sheltered and evacuated before radioactive release begins for many of the SOARCA scenarios. The timeline notes key accident progression and emergency response events.

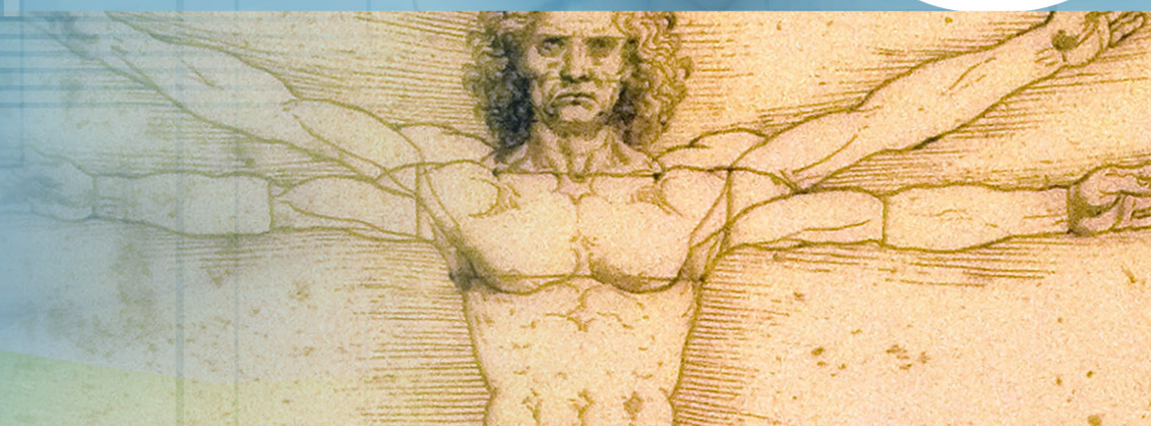
In each analyzed scenario, the plants follow their stated emergency response plans and promptly notify offsite authorities who activate their emergency notification systems (sirens) and direct the public to evacuate.

Figure 5.1 Example of Evacuation Timing for Peach Bottom Unmitigated Long-Term Station Blackout Base Case (hours)



MODELING HEALTH EFFECTS

CHAPTER 6



This chapter describes the models to calculate health consequences for SOARCA scenarios that release radioactive materials to the environment.

How Is Radiation Measured?

Units that measure how much radioactive material decays over a period of time:

- Curie (Ci)
- Becquerel (Bq): 1 Bq = 2.7 x 10^{-11} Ci

Units that measure the effects of ionizing radiation on humans:

- rem
- Sievert (Sv): 1 Sv = 100 rem

More information about radiation and its health effects is at http://www.nrc.gov/about-nrc/radiation/rad-health-effects.html.

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

The team modeled the unmitigated scenarios' calculated releases and subsequent health consequences. Even in the unmitigated scenarios, modeling indicated that essentially no one would die from acute radiation exposure (due to the length of time for the accident to progress and the relatively small releases) and that there would be a very small possibility of long-term cancer fatality for an individual. This chapter provides an explanation and background information about how SOARCA modeled the health consequences.

HOW ARE HEALTH CONSEQUENCES REPORTED IN SOARCA?

Exposure to radiation can have a variety of different health effects depending on the specific type and intensity of exposure. In addition, radiation affects different people in different ways. Large, high-intensity exposures can cause acute health effects that range from nausea and skin reddening to death. In addition to acute health effects, radiation exposures are related to the occurrence of cancer later in life. The two types of health consequences reported in SOARCA are early fatalities from very large and intense exposures and fatalities that result from radiation-induced cancers.

Early Fatality Risk—Individual deaths that occur shortly (usually within a few weeks or months) after exposure to large doses of radiation. The report provides this number as the average individual risk of an early fatality. For scenarios analyzed, the early fatality risk is essentially zero.

Long-Term Cancer Fatality Risk—Cancer fatalities that occur years after exposure to radiation. This number represents the average individual risk of dying from cancer due to radiation exposure following the specific hypothesized severe accident scenario. For the scenarios analyzed, long-term cancer fatality risk is very small.

HOW ARE LONG-TERM CANCER FATALITY RISKS MODELED?

Modeling long-term cancer fatality risk is controversial because medical researchers disagree on the evidence that describes the adverse effects of low radiation doses. The SOARCA project used two long-term cancer fatality risk models to provide additional information on the effects of different modeling approaches on the potential range of health consequences:

Linear-no-threshold dose response model— This model is based on the conclusion that any amount of radiation dose (no matter how small) can incrementally increase cancer risk. It is a basic assumption used in many regulatory limits, including NRC's regulations and past assessments.*

Linear-with-threshold dose response model— To provide additional information on the potential range of health consequences, the SOARCA project calculated long-term cancer fatality risk assuming the linear-no-threshold model and a range of threshold or cutoff doses below which the cancer risk is not quantified. When comparing offsite consequence results for the linear-no-threshold model and linear-with-threshold model, these threshold values make the already small long-term cancer fatality risk values even smaller (by orders of magnitude in some cases).

SOARCA uses multiple dose threshold values, including:

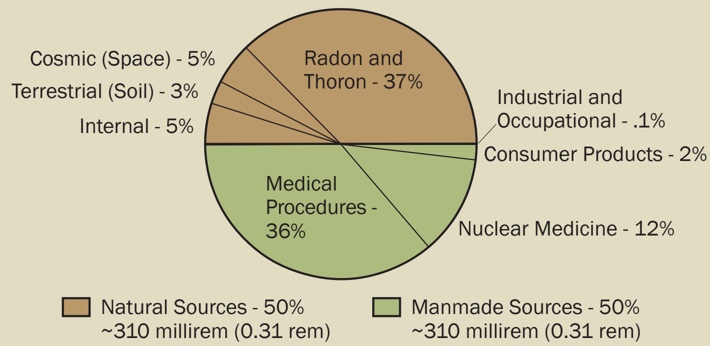
620 mrem per year—This represents the U.S. average individual background dose including medical exposures.

5 rem (5,000 mrem) per year with a 10 rem lifetime cap—This value was chosen based on the Health Physics Society position statement in "Radiation Risk in Perspective" (July 2010).

As a resident of the United States, how am I exposed to radiation?

SOARCA studies health effects in situations where a severe accident releases radiation to the public. To provide some perspective, people generally receive an average total dose of ionizing radiation of about 620 millirem per year. Of this total, the chart shows that natural sources of radiation account for about 50 percent and manmade sources account for the other 50 percent.

Sources of Radiation Exposure in the United States



Source: NCRP Report No.160(2009)
Full report is available on the NCRP Web site at www.NCRPpublications.org.

** Use of the linear no-threshold model for low radiation exposures (below 0.1 sievert or 10 rem) to project future long-term cancer fatality risk to individuals receiving such exposures is currently being debated within the scientific community. Many radiation protection organizations, such as the U.S. National Council on Radiation Protection and Measurements, the International Commission on Radiological Protection, the United Nations Scientific Committee on Exposure to Atomic Radiation, and the U.S. Health Physics Society, caution that there is considerable uncertainty when computing cancer deaths resulting from small additional exposures to large populations over many years and should only be done under explicit conditions such as in the SOARCA project or not at all.*

The MACCS code looks at atmospheric transport of radioactive material using a cloud, or plume, that travels in a straight line following the wind direction. This model of short-term and long-term dose accumulation includes several pathways: radiation from the plume (cloudshine), radiation from material that reaches the ground (groundshine); inhalation, deposition onto the skin, and food and water ingestion. The ingestion pathway was not used in the analyses reported here because uncontaminated food and water supplies are abundant within the United States, and it is unlikely that the public would eat radioactively contaminated food. The following dose pathways are included in the reported risk metrics:

- Cloudshine during plume passage
- Groundshine during the emergency and long-term phases from deposited aerosols
- Inhalation during plume passage and following plume passage from radioactive dust kicked up by weather or human and vehicle traffic. This dust factor covers both the emergency and long-term phases.

Figure 6.1 Transport Pathways of Radioactive Materials

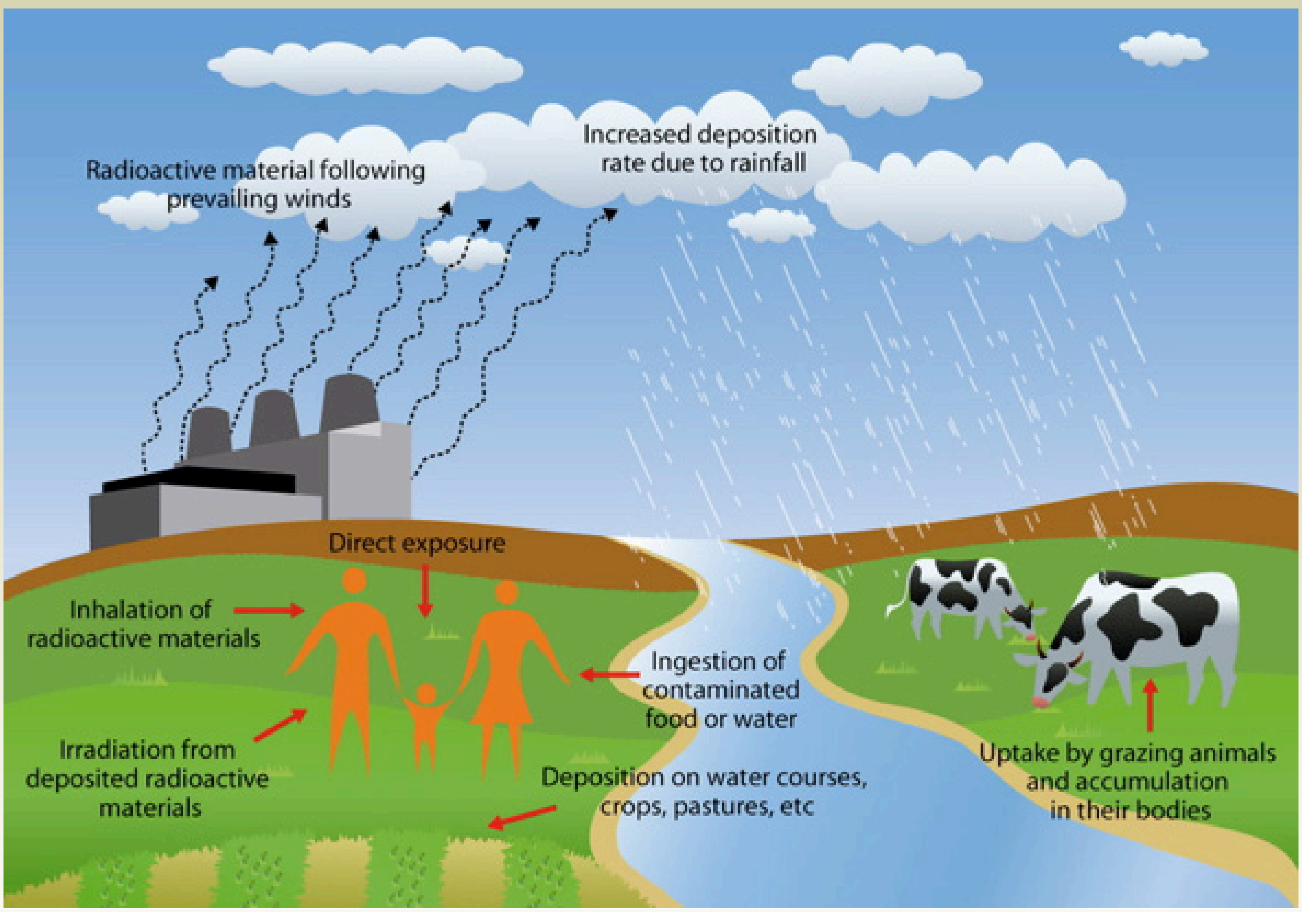
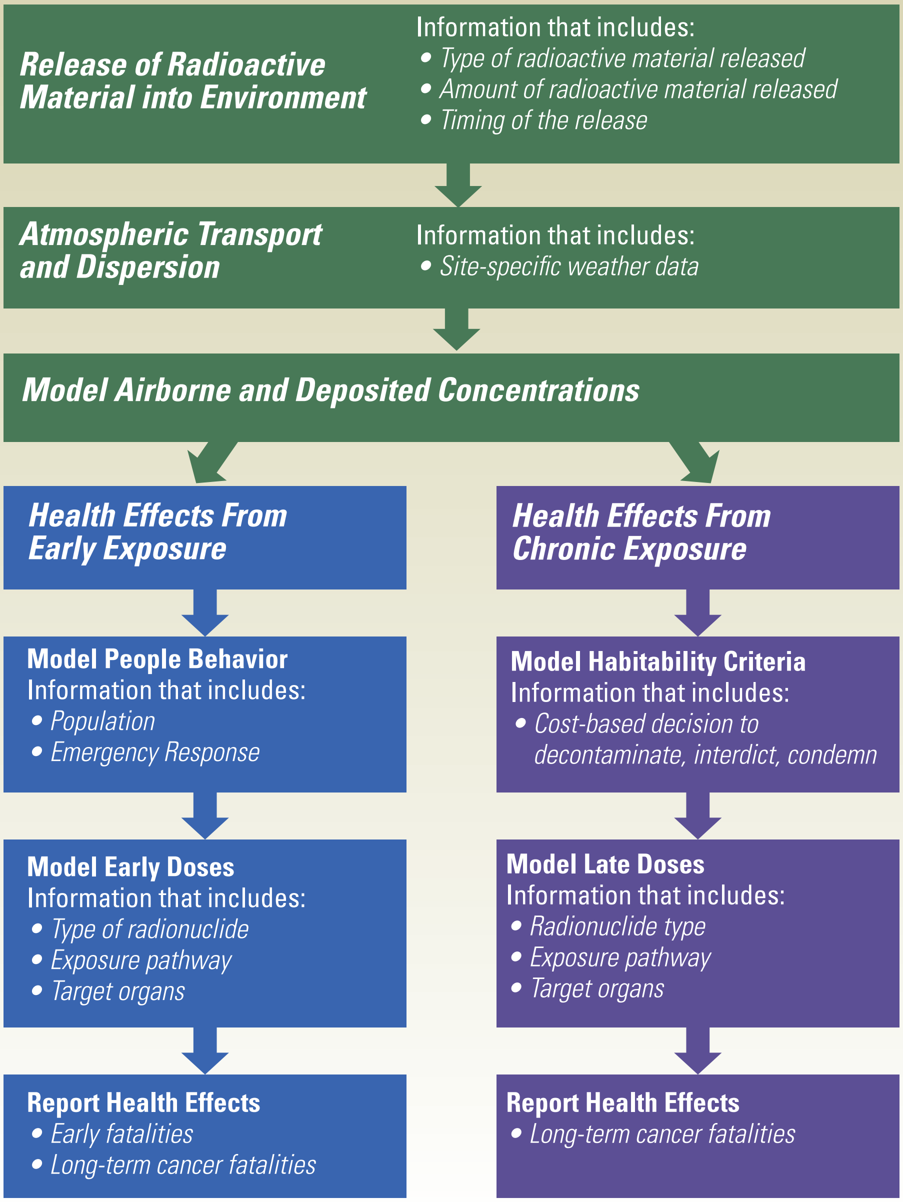


Figure 6.2 Information Used to Model Health Effects in SOARCA



WHAT INFORMATION IS INCLUDED IN THE MACCS MODELING?

In MACCS modeling we consider the following:

- When and at what rate the accident releases radioactive material into the environment (from MELCOR analysis described in Chapter 4)
- Protective measures (such as evacuation) taken by the offsite population (from the modeling of emergency plans described in Chapter 5)
- Site-specific weather data
- Downwind transport of the radioactive material released into the environment
- How each type of radionuclide will impact the body
- Radiation exposure of the offsite population and the health effects caused by this exposure

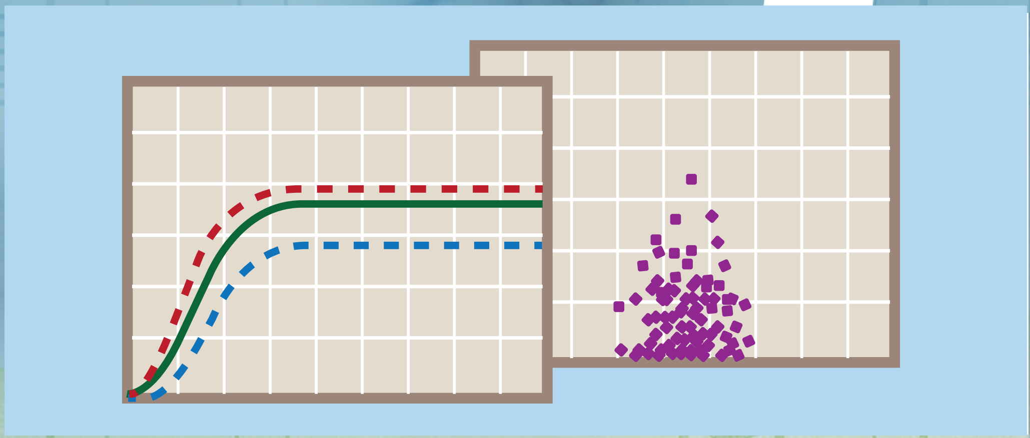
HOW ARE RADIOACTIVE MATERIALS MODELED TO MOVE DOWNWIND AND AFFECT THE POPULATION?

Radioactive materials are released from plant buildings as aerosol particles in a plume of steam and other gases. MACCS uses site-specific weather data to calculate the downwind concentration of radioactive material in the plume and the resulting population exposures and health effects. MACCS then applies a statistical model to calculate the average individual fatality risk as a result of the variability in the weather.

SOARCA modeled individual radiation exposure from inhaling the aerosol particles and by direct radiation from aerosol particles in the air and on the ground. A small portion of this exposure occurs during the early phase of the accident when the aerosol particles are being released from the plant buildings and while people are evacuating. Most of this is long-term exposure after land is decontaminated and people are allowed to return home. SOARCA modeled evacuees returning home based on guidance that outlines when it would be safe to do so. For the Surry and Sequoyah model, SOARCA uses the U.S. Environmental Protection Agency's "Manual of Protective Action Guides for Nuclear Incidents" to determine when the population can return to an area. For the Peach Bottom model, SOARCA uses Pennsylvania-specific criteria. This calculation also includes doses to the population in lightly contaminated areas where they were neither evacuated nor relocated. SOARCA did not model people who were exposed by eating food contaminated by aerosol particles because of the expected availability of uncontaminated food from other areas.

MODELING UNCERTAINTY

CHAPTER 7



This chapter explains uncertainty analysis, and how uncertainty is considered in computing the results.

What are the types of Uncertainty?

Incomplete knowledge about a system produces epistemic uncertainty. Increasing the body of knowledge available, such as performing experiments to test a certain parameter, can reduce this type of uncertainty. For example, if there is uncertainty on the pressure at which a component will fail, performing multiple experiments measuring the failure pressure can reduce the uncertainty. Inherently random uncertainty is known as aleatory uncertainty. No amount of study will predict the future behavior. An example of aleatory uncertainty is the weather at the time of an accident.

SOARCA uses complex computer codes and detailed models to simulate the accident progression and radioactive release (MELCOR) and the resulting offsite consequences (MACCS) of the postulated scenarios used for the SOARCA analyses. These models are based upon the best data available at the time the study was undertaken, however uncertainties remain in the behavior and definition of some parameters. An uncertainty analysis was performed for an accident scenario for each of the SOARCA pilot plants to determine the effect of uncertainties on the conclusions.

WHAT IS AN UNCERTAINTY ANALYSIS?

An uncertainty analysis tries to describe how changes in important input parameters affect a study's results. Understanding how the results scatter when the input parameters are varied can increase our confidence in the results. We can either lack knowledge about the parameters or we can consider randomness in the properties we are interested in. When we lack knowledge about some properties, we can increase the knowledge available and reduce the uncertainty by performing measurements or calculations. Random uncertainty will always exist.

SOARCA uses the "Monte Carlo" calculation technique in the uncertainty analysis. "Monte Carlo" calculation means that many calculations are run and parameters are varied in a defined range. A "probability distribution function" reflects how likely a value is to occur. The Monte Carlo technique samples these distributions to generate outputs.

SOARCA determined probability distributions for uncertain parameters. The probability distributions were sampled to pick values for uncertain parameters in a particular MELCOR and MACCS calculation, also referred to as a "realization." Each realization yields a unique set of outputs due to a particular radiological release. Combining the output of all the realizations reflects the range of possible results due to the uncertainty in the input parameters used.

WHAT ACCIDENT SCENARIOS WERE CONSIDERED?

A specific accident scenario was selected for each SOARCA uncertainty analysis to gain insights.

Peach Bottom Uncertainty Analysis—The Peach Bottom uncertainty analysis was a first-of-a-kind uncertainty analysis. It provided insights into the effect of uncertainties on the results and demonstrated the method. This analysis examined the Peach Bottom Unmitigated Long-Term Station Blackout accident scenario to study parameters such as the importance of back-up battery life.

Surry Uncertainty Analysis—The Surry uncertainty analysis examined the Surry Unmitigated Short-Term Station Blackout accident scenario. This scenario was chosen because it occurs relatively quickly. This scenario also included the possibility of an Induced Steam Generator Tube Rupture, which involves radionuclides bypassing the containment into the auxiliary building.

Sequoyah Uncertainty Analysis—The Sequoyah uncertainty analysis focused on the effect of hydrogen generation on the plant's containment. This study included unique parameters for the ice condenser containment used at Sequoyah. The Sequoyah analysis examined an Unmitigated Short-Term Station Blackout scenario. The study examined the effect of possible hydrogen build-up.

WHAT PARAMETERS WERE VARIED?

SOARCA chose parameters to study the effect of uncertainty in what is known about the parameters. Some random uncertainties were modeled, such as the effect of weather on the results. The uncertain parameters, and associated probability distributions, were chosen based on available data or expert opinion.

The results of previous studies informed parameter selection for later work. The Peach Bottom Analysis included the uncertainty of back-up battery life. No back-up batteries were available for the Surry and Sequoyah studies. The Surry analysis included uncertainty on the amount and type of radioactive material in the reactor core when the accident happens, as well as detailed modeling of the safety valves. The Surry analysis also included the uncertainty of an induced steam generator tube rupture. The Sequoyah study relied on the Surry analysis results and focused on the uncertainty regarding how the ice condenser containment functions during the accident.

WHAT IS THE OUTPUT OF AN UNCERTAINTY ANALYSIS?

The uncertainty analysis results are studied to understand the variation in the output of chosen parameters. Two primary output parameters for these analyses are individual latent cancer fatality risk and individual early fatality risk to the offsite population. Another output parameter is the accident's cesium release, because cesium has the greatest potential impact on latent cancer fatality risk among the types of radioactive material released. Similarly, iodine release is an output parameter of interest because iodine has the greatest potential impact on early fatality risk among the types of radioactive material released. Examining how the results change over time generate "horsetail" charts that display all the results for a particular output. These horsetails demonstrate how the output changes due to the inputs.

The results of the uncertainty analyses are included in Chapter 4, Table 4.1, and confirmed conclusions from the original SOARCA analyses, which are presented in Chapter 8.

How is the SOARCA Uncertainty Analysis Performed?

The NRC developed detailed, site-specific, MELCOR and MACCS computer models for the study.

The SOARCA team identified uncertain parameters and their possible values, called cumulative distribution functions.

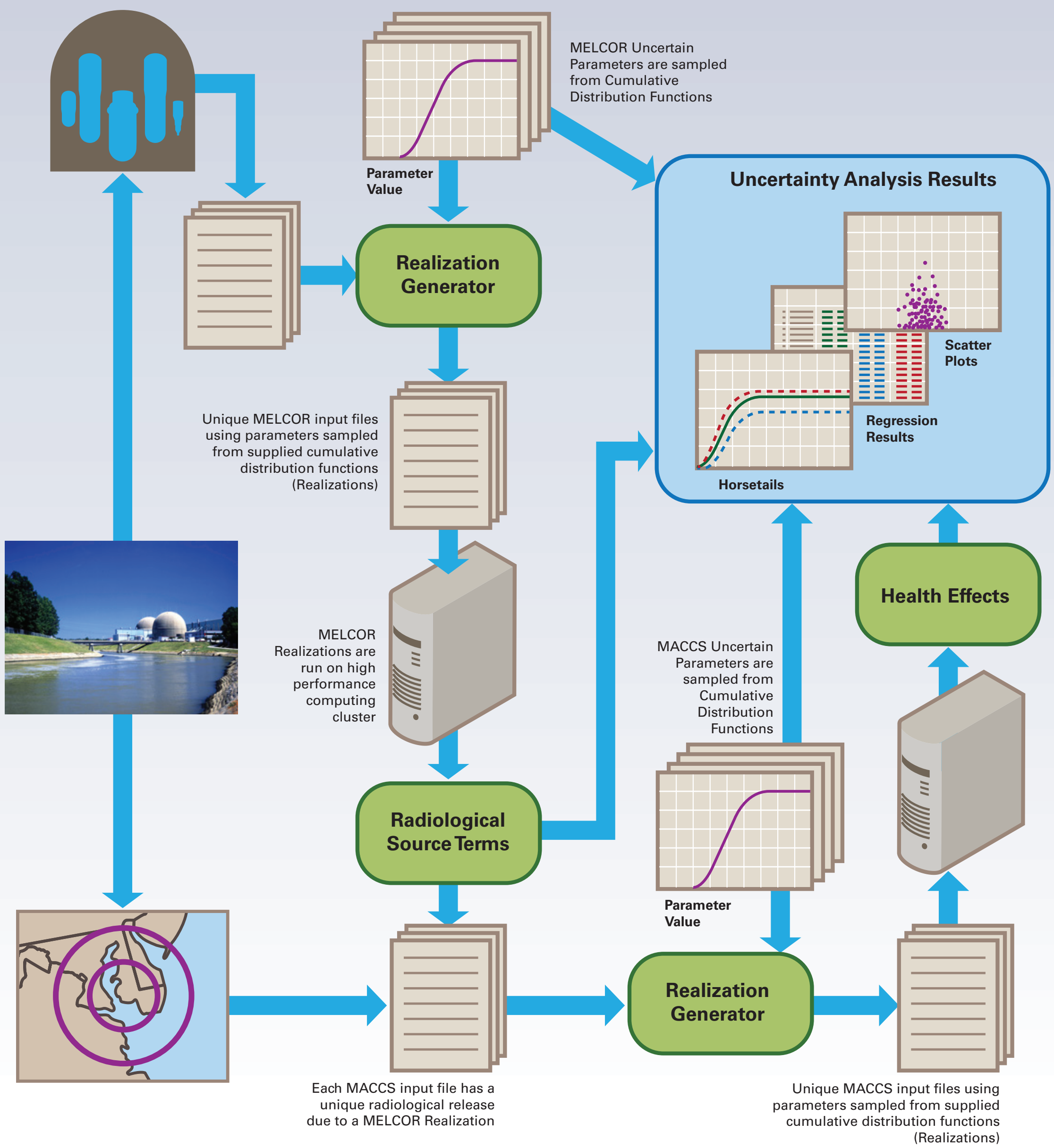
The cumulative distribution functions that could affect accident progression are sampled and combined with the base MELCOR input file to generate an appropriate range of input “realizations.”

The MELCOR realizations are run through a high performance computing cluster, or multiple personal computers, to calculate a range of radioactive material releases that become data for the MACCS input files.

The cumulative distribution functions that influence off-site consequences are sampled and combined with the unique MACCS input files to generate offsite realizations.

The MACCS realizations are run through a high performance computing cluster, or multiple personal computers, to calculate health effects.

The health effects and radiological releases are analyzed, using the sampled parameters, to generate “horsetails”, scatter plots, and regression results to quantify the uncertainty.



RESULTS AND CONCLUSIONS

CHAPTER 8

This chapter summarizes the results and conclusions from the SOARCA research project.

The SOARCA results demonstrate the potential benefits of the mitigation measures analyzed in this project. SOARCA shows that successful mitigation either prevents core damage or prevents, delays, or reduces offsite health consequences. In addition, the SOARCA team ran scenarios that demonstrate the consequences if certain mitigation measures are not successful. The unmitigated scenario results presented in this chapter demonstrate that, even in these cases, the public health consequences are very low.

The SOARCA project's offsite consequence analyses focused on radiation-induced fatality risks. However, it should be noted, that severe reactor accidents cause other types of offsite consequences which were not calculated in SOARCA. These include non-fatal cancers, displacement of population, economic losses, land contamination, as well as non-radiological health impacts. The Fukushima accident in Japan showed that severe nuclear accidents can have substantial offsite consequences even if they don't cause radiation-induced fatalities. Ofter NRC consequence analyses have evolved to include a more expanded set of the different types of offsite consequences that can be modeled with MACCS. For example, NRC's technical basis analyses (NUREG-2206) supporting the Containment Protection and Release Reduction rulemaking calculated population dose, economic costs, land contamination, and population subject to long-term protective actions and NRC's spent fuel pool consequence study (NUREG-2161) calculated collective dose, economic costs, land interdiction, and long-term displaced individuals.

WHAT ARE THE RESULTS OF THE MITIGATED SCENARIOS?

All mitigated cases of Peach Bottom and Surry SOARCA scenarios, except for one, result in prevention of core damage and/or no offsite release of radioactive material. The only mitigated case still leading to an offsite release was the Surry thermally induced steam generator tube rupture. In this case, mitigation is still beneficial in that it keeps most radioactive material inside containment and delays the onset of containment failure by about 2 days. For the Sequoyah analyses we only consider hydrogen igniters after core damage. The Sequoyah results show that early containment failure caused by hydrogen burns can be eliminated if igniters are operational within 3 hours.

As a result, the mitigated scenarios show zero risk of early fatalities from radiation exposure and result in either zero risk or very small risk of a long-term cancer fatality for an individual.

WHAT ARE THE RESULTS OF UNMITIGATED SCENARIOS?

The unmitigated scenarios result in essentially zero risk of early fatality for an individual. Although these unmitigated scenarios result in core damage and release of radioactive material to the environment, the release is often delayed, which allows the population to take protective actions (including evacuation and sheltering). Therefore, the public would not be exposed to dangerous amounts of radioactive material. This result holds even when uncertainties are considered – all three uncertainty analyses continued to show essentially zero risk of early fatalities.

For the unmitigated scenarios, the individual risk of a long-term cancer fatality is calculated to be very small—regardless of which distance interval (e.g., 0-10 miles, 0-20 miles, ... 0-50 miles) is considered. This result holds even when uncertainties are considered. Table 8.1summarizes the results based on the linear-no-threshold dose response model for estimating the risk for individuals located within 10 miles of each plant.

WHAT DO SOARCA RESULTS INDICATE ABOUT CONSEQUENCES OF SEVERE ACCIDENTS?

SOARCA results, while specific to Peach Bottom, Surry and Sequoyah, may be generally applicable to plants with similar designs. Additional work would be needed to confirm this, however, since differences exist in plant-specific designs, procedures, and emergency response characteristics. The SOARCA results for the three plants analyzed are as follows.

- When operators are successful in using available on-site equipment during the accidents analyzed in SOARCA, they can prevent reactor fuel from melting, or delay or reduce releases of radioactive material to the environment.
- SOARCA analyses indicate that all modeled accident scenarios, even if operators are unsuccessful in stopping the accident, progress more slowly and release smaller amounts of radioactive material than calculated in earlier studies.

Table 8.1 SOARCA Results: Mitigated and Unmitigated Cases

	About how likely is the accident to occur?	About what is the annual average risk[a] of a long-term cancer fatality for this scenario for an individual located within 10 miles of the plant?		
		Mitigated Case	Unmitigated Case	Approximate Range of Uncertainty[d]
Peach Bottom LTSBO	1 event in 300,000 reactor years	zero	1 in 3 billion	1 in 1 billion to 1 in 11 billion
Peach Bottom STSBO	1 event in 3 million reactor years	zero	1 in 20 billion	N/A
Surry LTSBO	1 event in 50,000 reactor years	zero	1 in 1 billion	N/A
Surry STSBO	1 event in 500,000 reactor years	zero[b]	1 in 6 billion	1 in 3 billion to 1 in 7 billion
Surry SGTR	1 event in 3 million reactor years	1 in 10 billion	1 in 10 billion	N/A
Surry ISLOCA	1 event in 30 million reactor years	zero	1 in 100 billion	N/A
Sequoyah LTSBO	1 event in 100,000 reactor years	zero[c]	1 in 200 million	N/A
Sequoyah STSBO	1 event in 500,000 reactor years	zero[c]	1 in 6 billion	1 in 3 billion to 1 in 50 trillion

a Estimated risks below 1 in 10 million reactor years should be viewed with caution because of the potential impact of events not studied in the analyses and the inherent uncertainty in very small calculated numbers.

b For the mitigated Surry STSBO, the reactor vessel would fail; however, the containment would not fail until about 66 hours after the blackout. A review of available resources and emergency plans shows that adequate mitigation measures could be brought onsite within 24 hours and connected and functioning within 48 hours. Therefore, 66 hours would allow time for mitigation via equipment brought to the site from offsite, and this mitigation would avert containment failure such that radioactive material would not be released to the environment.

c Although not explicitly modeled in the Sequoyah SOARCA, the response is expected to be similar to the mitigated Surry SOARCA assuming backup generators and pumps are available to restore core cooling.

d Values shown represent the 5th - 95th percentile range, for uncertainty in accident progression and offsite consequences. SOARCA did not evaluate uncertainty in accident frequency. Uncertainty analyses were performed for these 3 scenarios only.

- As a result, public health consequences from severe nuclear power plant accidents modeled in SOARCA are smaller than previously calculated.
- The delayed releases calculated provide more time for emergency response actions such as evacuating and sheltering for affected populations. For the scenarios analyzed, SOARCA shows that emergency response programs, if implemented as planned and practiced, reduce the risk of public health consequences.
- Both mitigated (10 CFR 50.54(hh)(2) operator actions are successful) and unmitigated (10 CFR 50.54(hh)(2) operator actions are unsuccessful) cases of all modeled severe accident scenarios in SOARCA cause essentially no risk of death during or shortly after the accident.
- SOARCA's calculated longer term cancer fatality risks for the accident scenarios analyzed are millions of times lower than the general U.S. cancer fatality risk.

HOW DO SOARCA RESULTS COMPARE TO PAST STUDIES?

The SOARCA offsite consequence calculations are generally smaller than reported in earlier studies. To provide perspective between SOARCA results and the more conservative estimates of severe reactor accident outcomes found in earlier NRC publications, SOARCA results are compared to the results of one of these previous publications: NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," commonly referred to as the 1982 Siting Study. SOARCA calculated essentially zero early fatality risk for the three sites. The exact basis for long-term cancer fatality results in the 1982 Siting Study could not be recovered. The 1982 Siting Study's computer code (CRAC2) is no longer available and some of the models and modeling choices used in that study could not be reconstructed. Therefore, the SOARCA team compared SOARCA results with the 1982 Siting Study results by replacing the SOARCA source term with the larger source term (SST1) assumed in the 1982 Siting Study. -The long-term cancer fatality calculations based on the 1982 Siting Study SST1 source term are higher than the long-term cancer fatality calculations for SOARCA scenarios, however the difference diminishes when considering larger areas out to a distance of 50 miles from the plant because in both studies, large populations are assumed to be exposed to small annual doses when protective actions are lifted and people return home after the accident.

HOW DO SOARCA RESULTS COMPARE TO THE OVERALL U.S. CANCER RISKS?

The SOARCA analyses for the three plants show essentially zero individual early fatality risk and a very low individual risk of fatal cancer for the population close to the plant. Even for the Sequoyah STSBO and the Surry SGTR variations leading to early containment failure in which the release to the environment begins prior to the completion of the EPZ evacuation, there is essentially zero individual early fatality risk and the individual cancer fatality risk to an individual within ten miles of the plant is very low – a lifetime risk of fatal cancer following an accident of approximately 1 in 1,000. When the very low likelihood of the accident occurring is also considered, which has been estimated to be on the order of one per several hundred thousand years of reactor operation, the overall risk is even much lower. For comparison, the average lifetime risk of fatal cancer to a member of the US population is approximately 1 in 5.

GLOSSARY AND REFERENCES

GLOSSARY

Acute health effects—Health effects which occur within two months of exposure

Advisory Committee on Reactor Safeguards (ACRS)—The ACRS is an independent review committee that advises the Commission, independent of the NRC technical staff, regarding the licensing and operation of reactor facilities and related safety issues, the adequacy of proposed reactor safety standards, technical and policy issues related to the licensing of new reactor designs, and other matters referred to it by the Commission.

Boiling Water Reactor—In a commercial boiling-water reactor, the reactor core creates heat and a single loop both delivers steam to the turbine generator 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 that can be powered by onsite diesel generators. Other safety systems, such as the containment building air coolers, also need electric power.

Containment Structure—An enclosure around a nuclear reactor to confine radioactive material that otherwise might be released to the atmosphere in the event of an accident. Pressurized-water reactor large dry containments are usually cylindrical with a dome- shaped top and made of steel-reinforced concrete and a steel liner. Ice condenser containments are cylindrical with a dome-shaped top made of steel and surrounded by a concrete shield building.

Coolant—A substance circulated through a nuclear reactor to remove or transfer heat. All commercial nuclear reactors in the United States use water.

Core Damage—Events that heat up the reactor core to the point at which fuel damage is anticipated or the drying out and heatup of the reactor core to the point at which prolonged oxidation and severe fuel damage lead 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 heat up to the point at which it would be damaged and potentially melt.

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

Emergency Operating Procedures (EOPs)—Plant-specific procedures containing instructions for operating staff to implement preventive measures for managing accidents.

Emergency Planning Zones (EPZ)—The 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 each site 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. The plume exposure EPZ extends about 10 miles from the plant, and the ingestion EPZ extends about 50 miles from the plant.

Evacuation Time Estimate (ETE)—The estimated time to mobilize and evacuate the public from a defined area. The ETE considers residents of the EPZ, transients, people visiting but not living within the EPZ, and special facilities including schools.

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.

Ingestion Pathway—The potential routes for radionuclides from various sources to enter water, the food chain, or get into a person's mouth in day-to-day activities.

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

MACCS—A general-purpose computer code for estimating offsite impacts following release of radioactive material. MACCS is applicable to diverse reactor and nonreactor situations. It considers atmospheric transport and dispersion under time-variable weather conditions, short- and long-term mitigation actions, and exposure pathways to determine doses, health effects, economic costs, and other types of consequences.

MELCOR—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 performed by plant operators to prevent core damage and/or the release of radioactive material.

Pressurized Water Reactor—In a 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 converts the water into steam in a secondary loop to drive the turbine generator to produce electricity.

Probabilistic Risk Assessment—A method to calculate risk by assessing both the probability of an event and its consequences. This procedure involves asking a series of three questions called the "risk triplet:" (1) What can go wrong? (2) How likely is it? (3) What would be the consequences?

Radiation—Energy that travels in the form of waves or high-speed particles. 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 Core—The central portion of a nuclear reactor which contains the fuel assemblies, moderator, neutron poisons, control rods, and support structures. The reactor core is where fission takes place.

Reactor Fuel—Boiling-water reactors and pressurized-water reactors use ceramic pellets containing enriched uranium dioxide (UO_2). These pellets are stacked and sealed inside long, slender, zirconium metal-based alloy (Zircaloy) tubes to form fuel rods. Fuel rods are assembled into bundles called fuel assemblies that are loaded into the reactor core.

Reactor-Year—The operation of one nuclear reactor for 1 year.

Severe Accident— A severe accident may challenge safety systems beyond a nuclear power plant's design limits, potentially damaging or degrading the reactor core and its containment buildings.

Severe Accident Management Guidelines (SAMGs)—Guidelines that plants voluntarily put in place in the late 1990s to contain or reduce the impact of accidents that damage a reactor core.

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

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