Schaperow, Jason

From: Schaperow, Jason
Sent: Thursday, January 06, 2011 1:20 PM
To: Chang, Richard; Santiago, Patricia
Cc: Tinkler, Charles
Subject: SOARCA NUREG/BR
Attachments: SOARCANUREGBR0471---v2.5.7_04022010_JHS.doc

Attached is the latest version of the SOARCA NUREG/BR that I have. It includes my comments in red via the Track Changes feature in Word.

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PERSONAL STATEMENT FROM BRIAN SHERON
Director
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

My colleagues and I at the U. S. Nuclear Regulatory Commission focus on protecting your health and the environment from civilian uses of nuclear materials, including the consequences of potential accidents at nuclear power plants. We have a professional commitment to carrying out this mission—including this research project to estimate public health and safety consequences in the unlikely event of a commercial nuclear power plant accident more severe than that for which the plant was originally designed to handle.~~releasing radioactive material into the environment.~~ This project examined a set of important scenarios for two plants. The project~~'s results indicate that commercial nuclear power plants are designed, operated, and regulated to prevent accidents and their safety systems and procedures can protect the public should an accident occur. We believe this work~~ integrates the plant owners' efforts over the last 50 years (issue: why 50 years?) to improve their plant designs and operations and the NRC's development of rigorous inspection methods, operator training, and emergency preparedness. The project's results indicate that t~~T~~hese changes have increased overall nuclear power plant safety. Also, national and international programs about severe accidents ~~and health effects~~ have increased our understanding of severe accidents. This increased understanding has been incorporated into the tools that NRC used to perform this study. We invite you to read this brochure to understand how we modeled nuclear power plant accidents using state-of-the-art methods in order to understand the outcomes~~consequences~~ of these unlikely accidents. (issues: It is an overstatement to say that MELCOR calculations indicate plants are designed, operated, and regulated to prevent accidents. This paragraph says the objective is to calculate offsite consequences; however, the actual objective is to estimate realistic outcomes one of which may be no core damage.)

Note:
One of the objectives of SOARCA is to enable the NRC to communicate severe-accident-related aspects of nuclear safety to stakeholders including federal, state, and local authorities, licensees, and the general public. To achieve this objective, the SOARCA results are documented in a draft document, NUREG-1935 and NUREG/CR-7008 and 7009 that will be provided to the public for review and comment and public meetings are planned. Because the NUREG is not generally accessible to some stakeholders due to its length and technical jargon, this booklet was developed as a plain language summary of the methods, results, and conclusions.

(Recommendation: Consider combining the two paragraphs on this page into some sort of "Brochure Overview." Not clear we need a personal statement from an NRC manager.)

KEY INSIGHTS:

- The results of this project indicate that reactor safety has improved over the years as a result of efforts by the commercial nuclear power industry to improve plant design and operation and by NRC to develop improved regulations to enhance safety.
- For the scenarios examined, mitigation of a release to the public is likely because operators have enough time and equipment to take the necessary actions. In all but two scenarios, mitigating actions were able to prevent core damage.
- If accidents are assumed to proceed as unmitigated,~~For the scenarios examined~~, our analyses indicate that potential radiation releases would occur several hours later than earlier thought, and they would be substantially smaller; as a result, our best estimate of early fatalities from severe accidents at nuclear power plants would be far less than previously calculated. (issue: This bullet conflicts with the previous bullet.)
- If accidents are assumed to proceed as unmitigated, our analyses ~~The analyzed scenarios~~ predict that essentially no deaths from radiation exposure will occur within weeks following the accident and long-term cancer fatality risks are very low ~~for the unmitigated scenarios examined~~. (This bullet needs to explain what an unmitigated scenario is.)
- If accidents are assumed to proceed as unmitigated, t~~T~~he predicted~~SOARCA~~ individual long-term cancer risk values ~~for the selected scenarios~~ are much smaller than the NRC-established safety goal that "individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health." (issues: This bullet needs to be clarified that it is for the unmitigated version of the scenarios. The phrase "for the selected scenarios" is confusing.)

ACKNOWLEDGMENTS:

TABLE OF CONTENTS
[note that the page numbers refer to the PDF version 2.5.5]

Chapter 1
Project Overview
This chapter explains the purpose of the project and the overall process for determining the results.

WHAT IS THE PROJECT'S PURPOSE?
The U. S. Nuclear Regulatory Commission's (NRC's) State-of-the-Art Reactor Consequence Analyses (SOARCA) research project is designed to estimate the realistic outcomes of ~~severe~~ nuclear power plant accident scenarios more severe than that for which the plant was originally designed to handle.~~that could release radioactive material into the environment.~~ The project also evaluated and improved methods and models for realistically evaluating plant responses during severe accidents, including protective actions for the public (such as evacuation and sheltering), and the potential public health risk. The NRC performed this study, in part, to develop information about the effectiveness of methods for mitigating severe accidents at nuclear power plants to prevent or minimize harm to the public. The SOARCA study seeks to produce more realistic estimates of plant behavior during severe accidents, thereby improving understanding of consequences of a potential accident.

The NRC, industry, and international nuclear safety organizations have extensively researched plant response to hypothetical accidents that could damage the core and containment. This research has ~~significantly~~ improved the NRC's ability to analyze and predict how nuclear plant systems and operators will respond to severe accidents. During that same time, plant owners enhanced plant designs, emergency procedures, inspection programs, and operator training, with the goal of improving~~all of which have improved~~ plant safety(issue: this is an overstatement). Plant owners and local governments have also refined and improved emergency preparedness to further protect the public in the ~~highly~~ unlikely event of a severe accident.

The SOARCA team applied this accumulated research and incorporated plant enhancements to achieve a more realistic evaluation of the consequences from severe nuclear reactor accidents. This study is an in-depth analysis of two operating nuclear power plants. Results, while specific to the two nuclear plants selected, may be generally applicable for plants with similar designs. However, results may vary depending on individual plant designs and capabilities and effectiveness of emergency response for surrounding populations. Following completion of the study, the NRC will consider additional applications of its results.

WHAT ARE THE RESULTS OF SOARCA ANALYSES?
SOARCA's results indicate that U.S. commercial nuclear power plants are designed, operated, maintained, and regulated to prevent accidents and to protect the public. (issue: It is an overstatement to say that MELCOR calculations indicate plants are designed, operated, and regulated to prevent accidents.)
For more than four decades, utilities have improved their plant designs and operations including mitigation measures and emergency preparedness with a goal of improving~~. All of these changes have increased overall nuclear power~~ plant safety. (issue: this is an overstatement)~~Based on the analyzed scenarios for the two plants,~~(issue: This is too wordy.) insights derived from SOARCA include:

- Newly incorporated **mitigative measures**, when implemented according to NRC rules, can further prevent radioactive releases and protect the public. (issue: This statement conflicts with itself. It first says the mitigative measures are newly incorporated. Then it suggests they may not been implemented according to NRC rules.)
- If accidents are assumed to proceed as unmitigated, accidents develop more slowly than previously thought, resulting in a delayed release of radioactive material to the environment.
- For the unmitigated versions of the scenarios, t~~T~~he modeled **radiation** releases are delayed and smaller than in past studies.
- As a result, the individual risk of death from radiation exposure, due to severe nuclear power plant accidents, is much less that previously predicted. (issue: different language is used here than in the Key Insights on page 2.)
- The estimated risk of cancer death from a severe accident a nuclear power plant is small. (issue: does not say that this is for unmitigated scenarios.)
- Results, while specific to the two nuclear plants selected, may be generally applicable for plants with similar designs. However, results may vary depending on individual plant designs and capabilities and effectiveness of emergency response for surrounding populations.

WHAT IS THE HISTORICAL BACKGROUND OF SEVERE ACCIDENT RESEARCH?
The NRC has previously researched the probabilities and potential health consequences of severe accidents. The full SOARCA report, NUREG-1935, NUREG/CR-7008 and NUREG/CR-7009, contains details about some of these past studies. These past studies were based on existing plant descriptions and knowledge of how severe accidents would occur. However, we now know that the predictions from these studies are not useful for characterizing consequences and guiding public policy. Since the publication of the earlier studies, the NRC has participated in ~~several~~ research

programs to improve ~~the staff's~~ understanding of the release of radioactive material from a reactor core and heat transfer and fluid flow characteristics during severe accidents. The staff (issue: who is "the staff?" This is not plain English.)incorporated these research results into SOARCA's computer codes. Additionally, the SOARCA study used a more complete and detailed model of reactors, containment, and other buildings on site. Because SOARCA is based on decades (issue: differs from previous characterizations: 40 years, 50 years, etc.) of research and uses improved modeling tools, the study generates more realistic results than past studies.

HOW IS SOARCA STATE-OF-THE-ART?

SOARCA is state-of-the-art for three reasons~~The NRC has three main reasons for considering SOARCA a state-of-the-art project:~~ (issue: too wordy):(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, and (3) it uses modern computer capabilities with very detailed plant models. When the NRC developed the SOARCA plant models, the staff interviewed plant personnel and examined current plant configurations to incorporate each facility's most current design and operational information. This updated information includes:

- ~~The most current s~~(issue: too wordy)Severe accident mitigation strategies and procedures, including enhancements made in connection with security-related events.
- System design improvements and operating and maintenance enhancements that improve plant reliability.
- Improved offsite emergency response programs and equipment.

(issue: include in the brochure a short list of remaining conservatives)

WHAT PLANTS DID SOARCA STUDY?

SOARCA analyzed an example of each major type of operating U.S. nuclear reactors: a boiling-water reactor (BWR) and a pressurized-water reactor (PWR). The project team solicited volunteers from the nuclear industry to participate in the project, and the Peach Bottom Atomic Power Station in Pennsylvania and the Surry Power Station in Virginia volunteered and they are the focus of this report (These two sites were also part of earlier accident analyses). SOARCA analyzed one reactor at each site.(issue: so what?) The NRC's Advisory Committee on Reactor Safeguards and independent, external experts reviewed the methods and results of SOARCA. The staff will consider extending SOARCA to analyze additional examples of operating U.S. nuclear reactors. (Peer review does not belong in this paragraph.)

(issue: include in the brochure a short status of the project.)

HOW ARE SEVERE ACCIDENTS AND THEIR CONSEQUENCES MODELED?

The SOARCA project used sophisticated 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, operating history, and the impacts of emergency procedures, weather conditions, emergency planning, and population.

WHAT WERE THE STEPS OF THE PROJECT?

The SOARCA project took a step-by-step approach to analyze the potential consequences of the more likely severe accidents. The project team first decided it could learn more by rigorously and realistically quantifying a relatively few important events, rather than carrying out approximate modeling of many events. Therefore, the team selected a threshold to help select scenarios to analyze (Chapter 2 describes the selection process). Each scenario was analyzed twice. For the "mitigated case" version, the team modeled how plant operators successfully implemented emergency plans and mitigating actions. For the scenarios examined, NRC staff (issue: not plain English) believes mitigation of a release to the public is likely because operators have enough time and equipment to take the necessary actions. In order to understand the value of mitigating actions and provide a basis for comparing SOARCA results to past studies, the team also ran an "unmitigated case" scenario in which newly available equipment (issue: and in some cases, ample time was enough together with TSC, EOF) was not used as additional mitigating actions. These scenarios included an analysis of the release of radioactive material, emergency preparedness, and health consequences. ***Figure 1.2*** illustrates the reasoning of this overall approach.

(issue: say "table-top exercises" somewhere in the brochure)

GRAPHICS AND SIDE BOXES FOR CHAPTER 1:

Who is the Project Team?
The project team included engineers and scientists from the NRC and Sandia National Laboratories. The team's expertise included probabilistic risk assessment, heat transfer and fluid flow, onsite and offsite emergency response, 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.

How to Use this Brochure
Regulators and industry groups have been researching the consequences of severe nuclear power plant accidents since the 1950s. This booklet provides tools to help understand SOARCA's processes, terminology, and results of SOARCA. Here are some features that you can use:
-Colored boxes such as this one explain concepts
-Glossary in the appendix defines terms
-Side boxes provide historical information or explain relevant NRC regulations
-References in the appendix provide a list of information documents
If you are viewing this online:
-Bolded terms are linked to the glossary
-Gray, underlined phrases and URLs are linked to the NRC website

WHAT IS A SEVERE ACCIDENT?
It is an accident that may challenge safety systems at a level much higher than expected in the original design of the nuclear power plant facility. As such it can damage or degrade the reactor core and its containment buildings.

Historical Perspective: Three Mile Island and Chernobyl
Many people are familiar with the accidents that occurred at Three Mile Island and Chernobyl. Even though SOARCA did not examine these historical accidents, this brochure periodically provides specific information about these accidents so that readers can compare the results of this study to history.

The Three Mile Island accident in Pennsylvania on March 29, 1979, melted almost half the reactor core of Unit 2, and released contaminated water and radioactive material into the containment building. It was the most serious in U.S. commercial nuclear power plant operating history, even though it led to no deaths or injuries to plant workers or members of the nearby community. This accident brought about sweeping changes for nuclear power plants and heightened oversight by the NRC.

On April 26, 1986, an accident occurred at Unit 4 of the nuclear power station at Chernobyl, Ukraine, in the former USSR. The accident, caused by a sudden surge of power, destroyed the reactor and released massive amounts of radioactive material into the environment. About 30 emergency responders died in the first four 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. To date there is no strong evidence for radiation-induced increases of leukemia or solid cancer (other than thyroid cancer). The design of that reactor, which differed significantly from reactors operating in the U.S., made it vulnerable to such a severe accident.

The NRC Fact Sheets about Three Mile Island and Chernobyl Accidents are at http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/3mile-isle.html and http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/chernobyl-bg.html.

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

WHAT COMPUTER CODES ARE USED FOR SOARCA?
SOARCA uses two specialized computer codes to analyze severe accidents and offsite consequences. The first, MELCOR, calculates the timing and event progression of accidents using specific plant design parameters. The second, MACCS2, calculates the offsite consequences of an airborne release of radioactive material using site-specific information for the area and radiological release data from MELCOR.

Figure 1.1 Location of All Reactors in the U.S. (on page 5 of PDF version 2.5.5)

Figure 1.2 Flowchart of the SOARCA Process (on page 8 of PDF version 2.5.5)

Chapter 2
Progression of Accident Scenarios

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

WHAT ARE DIFFERENCES BETWEEN REACTOR TYPES?
Figures 2.1 and ***2.2*** describe the differences between the two U.S. commercial nuclear power plant types: boiling-water reactors (BWR) and pressurized-water reactors (PWR). Within these two general categories, there are design variations at different sites. Whereas both types of reactors generate electricity, different types of accidents can develop due to the design differences at the plants.

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

Core damage occurs when accident conditions heat up the reactor core beyond the point at which fuel rods begin to rupture. Extensive core damage could lead to core melt, the process of reactor fuel melting and settling at the bottom of the reactor vessel or, if the accident is even more severe, the radioactive material causes the bottom of the reactor vessel to rupture.

The team used site-specific probability information to determine whether an accident scenario met the threshold for consideration. This specific information—data about each site's unique design, safety systems, components, and emergency plans—was used to determine the likelihood of the scenarios leading to core damage (also called core damage frequency). SOARCA's accident scenarios had to have a greater than one in a million reactor years (issue: reactor-years is not plain English) probability or greater than one in ten million reactor years probability for accidents that may bypass containment features. The team also considered scenarios that may have lower likelihoods than the threshold, but potentially higher consequences.

Occasionally, the question is posed, "What is the consequence of a worst case accident?" While it is completely understandable why such a question might be asked, it is important to understand that reactors are designed with extraordinary safety measures including multiple barriers (defense in depth), and redundant and diverse safety systems incorporating both active and passive features. We purposely did not select scenarios on the basis of being "worst case scenarios" because using this type of criteria leads to a selection of events that are incredibly unlikely. We believe that concern over the severity of the event must be balanced with consideration of its likelihood and that "worst case scenarios" are appropriately excluded if they are not reasonably credible. Never-the-less, the selected scenarios are quite severe and unlikely because of the safety design of the plants. Furthermore, the NRC will continue to examine safety aspects of nuclear reactors, including conditions that might lead to severe accidents.

WHAT WERE THE SOARCA SCENARIOS?
For both sites (Peach Bottom and Surry) the team modeled the following scenarios called station blackouts, which were assumed to be caused by an earthquake more severe than the plant was designed to withstand. Other events, such as grid failure, floods, or fire, can trigger these scenarios; however, SOARCA modeled the scenarios that presented the most severe challenge to the plant operators and have the highest probability.

> ***Long-Term Station Blackout*** (LTSBO)—In this scenario, the station (issue: changed from "plant" to "station") loses all alternating current power sources but battery backups operate safety systems for about 4 to-6 hours until the batteries are exhausted.
>
> ***Short-Term Station Blackout*** (STSBO)—In this scenario, the site loses all power, even the batteries, and therefore all of the safety systems immediatelybecome ~~quickly~~ inoperable in the "short term."

Additionally, the team identified two scenarios for the PWR design at Surry in which radioactive material can reach the environment by bypassing containment features.

> ***Interfacing-Systems Loss-of-Coolant Accident*** (ISLOCA)—In this scenario, a random failure of check valves causes a rupture in the low-pressure system piping outside the containment.
> ***Thermally Induced Steam Generator Tube Rupture*** (TISGTR)—This scenario is a low-probability variation of the short-term station blackout. While the core is overheating and boiling off the available water, extremely hot steam and hydrogen flow out and cause a steam generator tube to rupture from the heat.

HOW WERE THE ACCIDENTS MODELED?

The SOARCA team realistically modeled the accident scenarios and their potential to damage the core by gathering detailed information about each of the two 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 additional buildings on the plant site.

The ~~state-of-the-art~~ MELCOR computer code (issue: "computer code" here and "program" elsewhere) 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 the features of the accident.
- How the reactor core behaves as it heats up.
- How the fuel itself, the reactor piping, and the containment building behave under extremely high temperatures.
- Whether radioactive material reaches the environment, and if so, how it occurs and how much material is released.

This preceding list of information is based on the plant's design and physical safety systems. Additionally, nuclear plants ~~also~~ (issue: "additionally" and "also" in the same sentence) have a series of redundant and diverse safety measures to back up the designed safety systems. **Chapter 3** discusses how the SOARCA project models the mitigating actions that can prevent or mitigate the release of radioactive material and ultimately protect the public. If a scenario was modeled to produce a release of radioactive material, the team used another computer code to estimate 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 likelihoods that range from two events~~one accident~~ in 100,000 reactor-years (issue: the number here is wrong; it should be two)to one accident in 10 million reactor-years. (issue: reactor-year is not plain English) Table 2.3 shows the likelihoods for each scenario in order of more likely scenarios to less likely scenarios. Even though the chances of these scenarios ever occurring are very small, these are the scenarios that probabilistic risk assessments have shown to be important to risk. SOARCA examines the effectiveness of actions to mitigate each accident, should one occur, and prevent radioactive material from reaching the public and the environment.

GRAPHICS AND SIDE BOXES FOR CHAPTER 2

Figure 2.1 Typical U.S. Boiling-Water Reactor
[Image is on page 10 of PDF Version 2.5.5]
Whereas a PWR has separate coolant loops for cooling the reactor and generating steam (issue: this seems like a strange way to start explaining a BWR), the BWR cools the core with water, which also helps moderate the fission chain reaction. Heat from nuclear fission in the reactor core converts the water to steam. The steam passes through a two-stage moisture separator in the top of the reactor to remove any water droplets before leaving the reactor via the steam line. The dry 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 1100 pounds per square inch (psi) pressure (issue: awkward to say "pressurized to 1100 psi pressure) (issue: gauge or absolute?) so it boils at about 550 °F. In comparison, there is no significant boiling allowed in a PWR because the primary coolant loop is maintained at about 2300 psi. A typical BWR core contains between 400 and 800 fuel assemblies. Each fuel assembly comprises 75 to 100 fuel rods.

Figure 2.2 Typical U.S. Pressurized-Water Reactor
[Image is on page 11 of PDF Version 2.5.5]
Whereas a BWR boils the reactor coolant water to generate steam, the 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 (approximately 2300 psi) to prevent it 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 (approximately 900 psi) to produce saturated steam at about 530°F. This steam travels through a two-stage moisture separator in the top of the steam generators and then enters the main steam line which routes it to the turbine generator. From the turbine generator, the exhausted steam enters the condenser which 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; each assembly contains 200 to 300 fuel rods in a 14x14 to 17x17 matrix.

Reactor pressure vessel head being removed from the vessel

What Is Probabilistic Risk Assessment (PRA)?
The NRC assesses risk by computing the probability of an event in relationship to its consequences. One way a risk value can be represented is with the equation:

Risk = Probability x Consequences

PRA determines risk by asking a series of three questions called the "risk triplet":
-What can go wrong?
-How likely is it?
-What would be the consequences?

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

NEED
PHOTO
HERE

PWR Steam Generator

Steam generator with cover removed showing the steam generator tubes

What Are NRC Regulations?
General Design Criteria
In 1971, the Atomic Energy Commission (the NRC's predecessor) published detailed design criteria for commercial U.S. power reactors which can be found in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."
Criteria about quality—These criteria set requirements for how structures, systems, and components (issue: the phrase "structures, systems, and components" is not plain English) must meet standards for safety performance and set requirements for how licensees' quality assurance/quality control programs must be maintained.
Criteria about protection—These criteria set requirements for how commercial reactors must provide multiple layers of protection against natural phenomena and fission products and must provide control of the reactivity process.
Criteria about design—These criteria set requirements for reactor containment, movement of fluids and fuel.
The NRC amends the general design criteria as necessary to reflect current research and operating experience. These criteria can be found in 10 CFR Part 50, Appendix A or online at http://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-appa.html#1_appa.

Historical Perspective: How did the Three Mile Island Accident Unfold?
The accident began on March 28, 1979, when the feedwater pumps stopped running, preventing the steam generators from removing heat. Immediately, the pressure in the nuclear portion of the plant began to increase. A pilot-operated relief valve automatically opened to prevent that pressure from becoming excessive. The valve should have reclosed when the pressure subsequently decreased, but it stuck open~~did not~~. Control room information failed to show that the valve was still open. As a result, cooling water poured out of the stuck-open valve and caused the core of the reactor to overheat to a point where about one-half of it melted. The reactor vessel remained intact, the containment building withstood the increased pressure, and small releases of radioactive material occurred through a building adjacent to the containment. The public and environment were largely protected from the effects of the accident. Even though this accident resulted in a very small radiation release, the NRC learned from this accident and imposed new regulations on the industry to increase safety. More details are available in the NRC Backgrounder on the Three Mile Island Accident: http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/3mile-isle.html.

Historical Perspective: How did the Chernobyl Accident Unfold?
The accident at the Chernobyl Nuclear Plant, Unit 4 in Ukraine (then Soviet Union) occurred on April 26, 1986. The series of events that led to this accident could not occur at U.S. commercial power reactors because U.S. reactors have different plant designs, broader shutdown margins, robust containment structures, and operational controls to protect them against the combination of lapses that led to the accident at Chernobyl. Chernobyl's operators ran an experiment that led to an uncontrollable and extremely rapid power spike in the nuclear reaction. Within seconds, the core temperature rose above 5000 degrees Fahrenheit, melting the core and causing a steam explosion that destroyed the core and tore open the reactor building. The accident immediately released large amounts of radioactive material into the air and caused several fires (some burned for 10 days). To stop the fire and prevent a criticality accident as well as any further substantial release of fission products, boron and sand were poured on the reactor from the air. In addition, the damaged unit was entombed in a temporary concrete "sarcophagus," to limit further release of radioactive material. Although the NRC has always acknowledged the possibility of major accidents, its regulatory requirements provide adequate protection, subject to continuing vigilance, including review of new information that may suggest weaknesses. Assessments in the light of Chernobyl have indicated that the causes of the accident have been adequately dealt with in the design of U.S. commercial reactors.

Table 2.3 Likelihoods of SOARCA Accident Scenarios

Reactor Site	Accident Scenario	Probability
Surry	Long Term Station Blackout	2 events in 100,000 reactor-years
Peach Bottom	Long Term Station Blackout	3 events in a million reactor-years
Surry	Short Term Station Blackout	2 events in a million reactor-years
Peach Bottom	Short Term Station Blackout	3 events in 10 million reactor-years
Surry [1]	Thermally-Induced Steam Generator Tube Rupture	4 events in 10 million reactor-years
Surry	Interfacing-Systems Loss of Coolant Accident	~~8~~3 (issue: number is wrong; it should be 3) events in 100 million reactor-years[2]

~~1. Boiling water reactors, like Peach Bottom, do not have steam generators and therefore could not have an accident scenario like the thermally induced steam generator tube rupture.~~ (issue: this footnote is confusing and should be deleted)

2. The Surry plant owner estimates the likelihood of the interfacing-systems loss of coolant accident to be 7 events in 10,000,000 reactor years.

Chapter 3
Actions to Mitigate Accidents

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

WHAT ARE THE MITIGATING ACTIONS?

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

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

Severe accident management guidelines— These are guidelines for mitigating accidents more severe than what the facility was originally designed to handle.

Security-related mitigative measures—These measures include plans and resources nuclear plants put in place to meet additional NRC requirements following the terrorist attacks on September 11, 2001, as well as new procedures for using existing or additional equipment under accident conditions.

HOW DOES THE NRC KNOW THESE MITIGATING ACTIONS WILL WORK?

The NRC requires its licensees to train and practice emergency operating procedures in the plant simulators that are present at each site. The NRC also inspects to ensure that plant owners ~~the licensees~~ have developed severe accident management guidelines and implemented the security-related mitigative measures to ensure that they have proper equipment, procedures, and training.

HOW ARE MITIGATING ACTIONS MODELED?

SOARCA is the first detailed analysis that quantifies the value of these mitigating actions. For each plant, two cases of each scenario are modeled.

Mitigated Case— In the first case, the SOARCA team modeled what would happen if the operators successfully carried out the mitigating actions. The project team accomplished this by holding table-top exercises with senior reactor operators and emergency response personnel at Peach Bottom and Surry to determine what actions would be taken to mitigate each scenario analyzed, including the time required to implement each action. Analyses of each scenario using the MELCOR code showed that each scenario could reasonably be mitigated with no core damage in most cases.

Unmitigated Case— In order to understand the value of mitigating actions and provide a basis for comparing SOARCA results to past studies, the team also ran an "unmitigated case" scenario in which newly available equipment (and for ISLOCA, the installed equipment, TSC, EOF were not credited) was not used as additional mitigating actions. These cases modeled the sequence of events that lead to core melt, release of radioactive materials, and consequences to the public. (issue: the following important insight is missing: for some events, the time to core damage was so great that core damage could be prevented without the newly available equipment)

WHAT IS THE TIMING OF MITIGATING ACTIONS?

Detailed MELCOR modeling demonstrated that plant operators had ample time during accident scenarios to perform the necessary emergency actions. **Figure 3.1** illustrates the timeline for the Peach Bottom long term station blackout scenario from the blackout until the release starts (in an unmitigated situation) and compares that with the mitigating actions timeline.

Figure 3.1 Timing of Accident Progression for Peach Bottom Long Term Station Blackout [on page 20 of PDF Version 2.5.5]

GRAPHICS AND SIDE BOXES FOR CHAPTER 3

Defense-in-Depth Philosophy
"Defense in depth" is NRC's approach to designing and operating nuclear facilities (issue: this says the NRC designs and operates nuclear facilities) that prevents and mitigates accidents that release radiation or hazardous materials. The key is creating multiple independent and redundant and diverse 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 access controls, physical barriers, 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)."

Historical Perspective: How did Emergency Operations Improve after Three Mile Island?
Following the Three Mile Island accident, the NRC imposed additional procedural and training requirements. The nuclear industry now trains its reactor operators on emergency situations using full-scale, plant-specific control room simulators. The NRC emergency operations center is continually staffed and ready to respond if an accident begins to develop. The NRC has inspectors permanently stationed at each plant and requires more information from plant owners to ensure they make correct safety decisions.

Historical Perspective: How Has Security Improved since 9/11?
The NRC increased its ~~already rigorous~~ (issue: too wordy) security requirements for nuclear power plants to protect against sabotage, terrorism, and other intentional attacks. The new requirements include:
-increased patrols.
-additional security posts.
-additional physical barriers.
-vehicle checks at further distance.
-enhanced coordination with law enforcement and military.
-more restrictive site access controls.
For more information: *www.nrc.gov/security/faq-911.html#2*

Chapter 4
Release of Radioactive Material

This chapter explains how the project modeled radioactive material releases to the environment and what information is used in the calculations.

The SOARCA models showed that mitigating actions prevent or minimize ~~or prevent~~ core damage and therefore minimize or prevent a release of radioactive material. For the scenarios examined, the SOARCA team believes that these mitigating actions are a reasonable expectation, but the team also modeled the scenarios as proceeding unmitigated, leading ultimately to a ~~hypothetical~~ release to the environment. The MELCOR computer code models the behavior of radioactive materials to the point that they exit containment.

WHICH RADIONUCLIDES DOES SOARCA MODEL?
SOARCA tries to be thorough in its consideration of radionuclides. In MELCOR, SOARCA considers decay heat from approximately 800 radioactive substances, or radionuclides, and organizes them by chemical similarity. Of these 800, the offsite consequences computer code (MACCS2) tracks approximately 60 individual radionuclides based on how long they remain radioactive, their biological importance, and how much is expected to be released from the core. (issue: so what?)

> **Cesium and Iodine**—These two radionuclide 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.
>
> **Noble gases**—These radionuclides (krypton and xenon) are chemically inert ~~very stable~~ and therefore unlikely to impact the human body, even though they are expected to be part of an accident release.
>
> **Other radionuclides**—MELCOR and MACCS2 consider the complete radiological inventory in the analysis, and consequence results in NUREG-1935 include health effects from all radionuclides released in the accident.

WHAT INFORMATION IS INCLUDED IN MELCOR MODELING?
- How physical and chemical processes influence the behavior of radioactive material while the core heats up.
- How the extremely high temperatures influence particles' behavior at the molecular level and their physical states (e.g., 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 sprays and fan coolers) impact the behavior of radioactive material to prevent release.
- When and at what rate the accident releases radioactive material into the environment.

HOW ARE RADIOACTIVE MATERIALS MODELED TO EXIT CONTAINMENT?
The following sections describe the timing of the movement of radioactive material while on site and when it is released to the environment. Figure 4.1 shows the percentage of the reactor core inventory of radioactive iodine and cesium released to the environment during the first 48 hours of the accident.

(issue: need to start new paragraph) Reactor fuel consists of pellets of uranium dioxide inside a fuel rod made from zirconium alloy (zircaloy). During reactor operation, some of the uranium atoms in the fuel pellets split, releasing energy and radiation. If fuel rods are not adequately cooled, heat generated by decaying radioactive isotopes may cause them to overheat, possibly causing ruptures that allow some of the radioactive isotopes to escape. In summary, when the physical barriers (the fuel rods, the reactor vessel, and the containment building) are breached through the core melt process, the radioactive material divides itself according to chemical and physical properties that it has. The following sections provide more detail about this for the SOARCA scenarios.

HOW DOES RADIOACTIVE MATERIAL BEHAVE DURING THE STATION BLACKOUT SCENARIOS?
As a result of an accident involving a sustained loss of cooling water, the heat released from the radioactive material within the fuel pellets can cause the zircaloy tubes to rupture. The heat causes some of the radioactive material (such as cesium and iodine) to change into a gaseous form and move into the reactor coolant system through the rupture. As the radioactive material moves away from the core in gaseous form, it cools off and forms very small particles which can then deposit onto the inside surfaces of the reactor coolant system.

The following text describes the movement of radioactive material for the two analyzed Peach Bottom station blackout events:

- *Behavior of radioactive material while the core is in the reactor vessel*: Radioactive material moves from the fuel into the bottom of the suppression pool. Some material deposits on reactor vessel and pipe ~~and valve~~ (issue: this is more accurate) surfaces on its way to the suppression pool; the rest is captured in the suppression pool as it bubbles up through the pool's 23 feet of water.
- *Behavior of radioactive material after the degraded core leaves the reactor vessel:* 20 hours after the LTSBO begins (8 hours for the STSBO), partially molten core material melts through the bottom head of the reactor vessel, lands on the containment floor, spreads across the floor, and contacts the steel containment shell, melting a hole through it. Some radioactive material flows into the reactor building through this hole. Some material is released through the reactor building and some of it reaches the environment via blowout panels atop the reactor building. Some material deposits on inside surfaces as it moves through the containment and the reactor building on the way to the environment. Most of the radioactive material remains inside the containment and reactor building. (issue: this is more accurate)

The following text describes the movement of radioactive material for the two analyzed Surry station blackout events:

- *Behavior of radioactive materials while the core is in the reactor vessel*: While the fuel is overheating, radioactive material flows into the containment building through a rupture in the reactor coolant system piping. Some material deposits on the inside surfaces of the reactor coolant system as it flows to the containment building. The remaining material deposits in the containment building.
- *Behavior of radioactive material after the degraded core leaves the reactor vessel*: 21 hours after the LTSBO begins (7 hours for the STSBO), partially molten core material melts through the bottom head of the reactor vessel and lands on the containment floor. Some radioactive material flows into the containment building where it deposits. Later (46 hours for the LTSBO and 26 hours for the STSBO), the pressure in the containment building reaches the building's limit. The containment's steel liner tears and the reinforced concrete cracks. Most of the radioactive material has deposited in the containment with little remaining airborne inside the containment building.

HOW DOES RADIOACTIVE MATERIAL BEHAVE DURING THE SURRY BYPASS SCENARIOS?

The ISLOCA scenario begins with the random failure of two valves in series, rupturing a pipe outside of the containment building, providing a path from the reactor core to the environment that bypasses containment.

- *Behavior of radioactive material while the core is in the reactor vessel*: The fuel begins to overheat about 10 hours after the scenario begins. When the overheating fuel is in the reactor vessel, some of the radioactive material flows from the fuel through the ruptured pipe and into the auxiliary building. Most of this radioactive material deposits in the auxiliary building, with a fraction of it entering the environment.
- *Behavior of the radioactive material after the degraded core leaves the reactor vessel*: 15 hours after the scenario begins, partially molten core material melts through the bottom head of the reactor vessel and lands on the containment floor. Most of the radioactive material remains inside the containment building.

The TISGTR scenario is a low-probability variation of the short term station blackout that includes a steam generator tube rupture during core heatup and degradation.

- *Behavior of radioactive material while the core is in the reactor vessel*: About 3.5 hours after the scenario begins, a steam generator tube ruptures—creating about a one-inch diameter hole. Minutes later, a reactor coolant system pipe also ruptures—creating about a two 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 into the environment (issue: need to verify this statement). After the pipe rupture, the radioactive material predominantly flows into and deposits in the containment.
- *Behavior of the radioactive material after the degraded core leaves the reactor vessel*: About 7.5 hours after the scenario begins, partially molten core material melts through the bottom head of the reactor vessel and lands on the containment floor. Some radioactive material flows from the fuel and deposits in the containment building. Most of the radioactive material remains inside the containment building.

GRAPHICS AND SIDE BOXES FOR CHAPTER 4

How Does Containment Work?
As part of the defense-in-depth philosophy, the NRC requires all currently operating reactors to have these three physical layers of containment that protect the public and environment from potential releases of radioactive material:
Containment Building—enclosure around a nuclear reactor to confine fission products that otherwise might be released to the atmosphere in the event of an accident.
Reactor Vessel— metal enclosure that holds the reactor core and the coolant.
Fuel Rods—long, slender tubes that hold fissionable material (fuel) for nuclear reactor use. Fuel rods are assembled into bundles which are loaded individually into the reactor core.
page 22 of version 5 shows how text and image are connected

Diagram of the components of a reactor fuel assembly.

Historical Perspective: What Were The Releases From The Three Mile Island and Chernobyl Accidents?
The accident at Three Mile Island resulted in a very small release of radioactive material because it had a strong containment. The Chernobyl release was much more severe because of the large power increase that occurred as part of the accident. This release resulted in ~~widespread~~ contamination in areas of Belarus, the Russian Federation, and Ukraine. This figure compares the iodine and cesium releases of these two accidents with the estimated releases of the SOARCA unmitigated scenarios. The table reports the releases in Curies (Ci) which are units that measure how much radioactive material decays over a period of time (Megacuries, MCi, are one million curies.)

The NRC Backgrounders (issue: "Backgrounder" is not a word) about Three Mile Island and Chernobyl Accidents are at: http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/3mile-isle.html and http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/chernobyl-bg.html.

	Release of radio-iodine (I-131)	Release of radio-cesium (Cs-137)
Historical Accidents		
Three Mile Island	< 20 Ci	No release
Chernobyl	7 MCi	2.3 MCi
SOARCA Unmitigated Scenarios		
Peach Bottom Long-Term Station Blackout	3 MCi	0.2 MCi
Peach Bottom Short-Term Station Blackout	9 MCi	0.2 MCi
Surry Long-Term Station Blackout	0.2 MCi	0.002 MCi
Surry Short-Term Station Blackout	0.4 MCi	0.005 MCi
Surry Interfacing Systems Loss-of-Coolant Accident	7 MCi	0.8 MCi
Surry Thermally Induced Steam Generator Tube Rupture	0.7 MCi	0.04 MCi

FIGURE 4.1 Percentage of Iodine and Cesium Released to the Environment During the First ~~-~~48 Hours of the Accident
This figure compares the percentage of iodine and cesium from the reactor core that each accident scenario releases. These releases are much smaller than estimated in earlier risk and consequence studies. Also note that these releases can begin as early as 3.5 hours (for Surry ISLOCA) to as late as 45 hours (for Surry LTSBO) and some of these releases develop over a period of time.
[the figure appears on page 25 of PDF Version 2.5.5—note that we will move the 10% down to the lowest tick mark of the y-axis]

Chapter 5
Modeling emergency plans

This chapter explains emergency planning and how this~~ese~~ was~~ere~~ (issue: grammar) modeled.

SOARCA's mitigated scenarios were able to prevent a release, or limit it to noble gases, so even if the community had to evacuate as a precaution, the public would be protected. For the scenarios examined, the SOARCA team believes the implementation of these measures is reasonable; however the team also modeled the unmitigated scenarios leading to a release and requiring an evacuation. The models tracked the dispersion of radioactive material and the effect of emergency plans. This chapter provides more information about how the SOARCA project modeled emergency plans during a severe accident. In most scenarios~~,~~ (issue: in what scenario did we not have effective evacuation?), the releases' delayed timing (even without mitigative actions) allowed effective evacuation of local populations.

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

On-site Objective—Stop the accident. The NRC requires the utilities to have onsite response that includes technical, maintenance, and management staff that can respond within the first hour of the start of the accident. Each year, the licensee trains and drills this capability and the NRC inspects it.0

Off-site Objective—Evacuate and/or shelter the local populations. The NRC requires utilities to have offsite response support from the local and State agencies. The Federal Emergency Management Agency inspects this capability every two years. Emergency planning zones (EPZs) help define where detailed protective action strategies would be used during an emergency. Every plant must have emergency action levels approved by the NRC that dictate when to declare an emergency, well before core melt or radiation release that results from a severe accident. This timing is designed to ensure that the plan will be implemented before the plant is in a serious state and that members of the public are well on their way to evacuation before the release 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 2000 U.S. Census and projected to 2005
- Evacuation time estimates from emergency plans

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. See Table 5.1 for a description of some of the groups. Other population groups modeled include people who evacuate on their own initiative prior to the evacuation order and people who do not evacuate.

WHAT DOES MODELING DEMONSTRATE ABOUT EMERGENCY PLANNING?
The MACCS2 computer code calculates the radiation dose to the public based on evacuating, sheltering, and returning to the area after the event. **Figure 5.2** (issue: this says Figure 5.2. where is Figure 5.1?) illustrates the modeled timing of the Peach Bottom LTSBO scenario and the timing of emergency response. Because the analyzed accident scenarios take several hours to begin releasing ~~release~~ radioactive material to the environment, this provides time for the population to evacuate before potential radiation exposure. Explicit consideration of seismic impacts on emergency response (e.g., loss of bridges, traffic signals and delayed notification) did not ~~significantly~~ impact risk predictions. Additionally, much of the estimated radiation exposure in the SOARCA scenarios is received after the population returns.

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

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

[appears on page 33 of PDF Version 2.5.5]

GRAPHICS AND SIDE BOXES FOR CHAPTER 5

Historical Perspective: What About the Emergency Plans at Three Mile Island?
The Three Mile Island accident revealed the need for better coordination between nuclear power plant operators and Federal, State, and local government emergency response organizations. Following the accident, the NRC improved its emergency preparedness regulations to require each nuclear power plant owner to submit the state and local government response plans for the plant's emergency planning zone (EPZ)~~.~~ (issue: the box below states there are 2 EPZs). The NRC requires each nuclear power plant to have an emergency response center for the utility to coordinate local, state, and federal activities. The NRC's 24/7 emergency operations center will provide support during emergencies.

What Are NRC Regulations for Emergency Plans?
~~**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 the 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.*

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 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 detailed planning for the EPZs enable emergency responders to take ad hoc actions beyond the EPZ if conditions should so dictate (issue: what is meant by "if conditions so dictate?").
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 from inhalation or ingestion of airborne radio-iodine ~~contamination~~.
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.

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

NRC Staff during an emergency preparedness drill
[Note that this was originally a side box, but since this chapter has so many figures and side boxes and the information was redundant, I simply kept the picture but propose removing the text.]

Chapter 6
Modeling Health Effects

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

~~Since the mitigated cases are able to prevent a release~~ (issue: mitigated TISGTR had a release) , T~~t~~he team modeled the unmitigated scenarios' calculated releases and subsequent health consequences to determine the mitigative actions' value. Even in the unmitigated scenarios, modeling, which indicated that there would be essentially no **early fatalities,** or deaths from radiation exposure (due to the length of time for the accident progression) and only a small possibility of **long-term cancer fatalities** (because of slowly developing and relatively small releases) (issue: grammar). These results indicate that commercial nuclear power plants are designed, operated, and regulated to prevent accidents and protect the public should a severe accident occur (issue: overstatement). Even in situations in which operators unsuccessfully enact (issue: awkward wording) emergency procedures, the risk of consequences to the public is extremely small. This chapter provides an explanation and background information about how SOARCA modeled the health consequences.

HOW ARE HEALTH CONSEQUENCES REPORTED IN SOARCA?
Health effects from radiation include early death (usually with a few weeks) from exposure to large doses of radiation and death resulting from radiation-induced cancers. Two types of health consequences are estimated and reported in SOARCA.

Early Fatality Risk— Individual deaths that occur shortly after exposure to large doses of radiation (usually within a few weeks). The report gives this number as the average individual likelihood of an early fatality.

Long-Term Cancer Fatality Risk—Cancer fatalities that occur years after exposure to radiation. The report gives this number based on whether a given scenario occurs and represents the average individual cancer mortality risk due to radiation exposure following the hypothesized accident (issue: too wordy).

HOW ARE LONG-TERM CANCER FATALITY RISKS MODELED?
Modeling long-term cancer fatality risk is controversial because of inconclusive ~~human~~ (issue: is it conclusive for other animals?) evidence regarding risk at low radiation doses. To provide additional information about how different modeling approaches may affect the potential range of health consequences, the SOARCA project developed long-term cancer fatality risk estimates using the following two approaches:

Linear-no-threshold model—This model suggests that any amount of radiation dose (no matter how small) can incrementally result in cancer. It is a basic assumption used in many regulatory limits ~~(including the NRC's regulations). This model is endorsed by the National Council on Radiation Protection and Measurements and the International Commission on Radiological Protection.~~ (issue: too wordy)

Truncation model—These models suggest that below certain doses, one should not quantify a cancer risk. S~~; s~~ome people~~experts~~ believe that there is a dose below which no cancer is induced. ~~and most experts agree that data on low-dose exposures is uncertain. The truncation model used in SOARCA is 10 mrem per year, representing a trivial individual dose (and is consistent with guidance from the International Commission on Radiological Protection). SOARCA performed two additional dose truncation calculations to compare the results to a range of expert opinions:~~ (Issues: "Expertise" does not seem to be relevant here. 10 mrem per year case does not need to be discussed in this brochure. Also, too wordy.)

620 mrem per year—represents the U. S. average individual background dose (including medical exposures)
5 rem per year with a 10 rem lifetime cap (issue: "lifetime cap" is not plain English). ~~This is the position of the Health Physics Society~~(issue: What is the technical basis for 5 rem/year? Is it occupational exposure? Is it A-bomb data? Is it the limit of statistical power?).

Using any of these models, our results show that latent cancer fatality risks are very low. This is because most to essentially all of the latent cancer fatality risk in our analyzed accidents is from doses less the average U.S. dose to individuals from background and medical sources (issue: Where did this conclusion come from? It is not in the Executive Summary.).

WHAT INFORMATION IS INCLUDED IN THE MACCS MODELING?

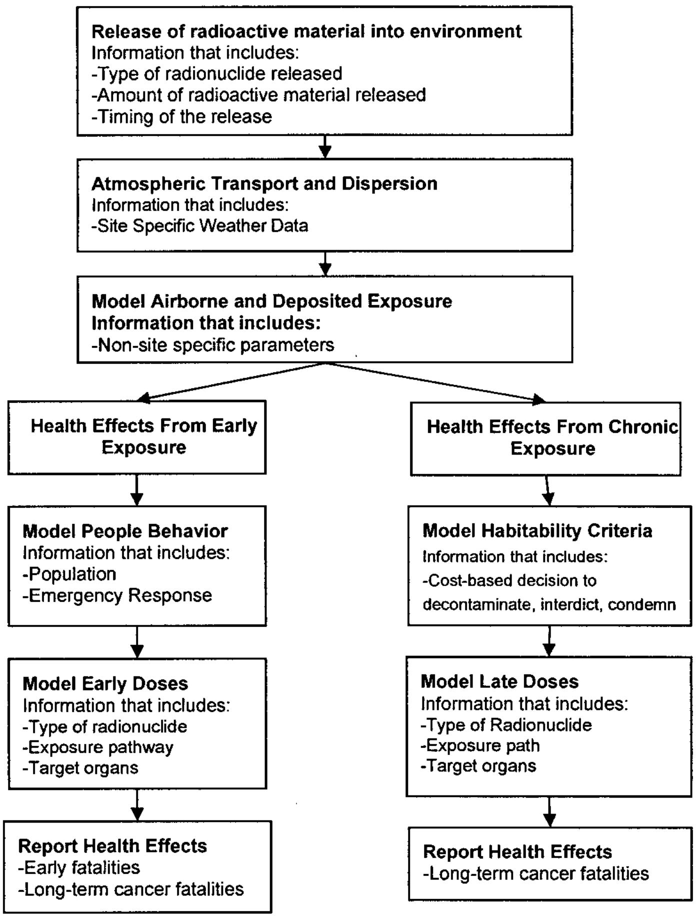
- When and at what rate the accident releases radioactive material into the environment. This information comes from the MELCOR analysis described in Chapter 4.
- Protective measures (such as evacuation) taken by the offsite population. This information comes from the modeling of emergency plans described in Chapter 5.
- Downwind transport of the radioactive material released into the environment.
- How each type of radionuclide will impact the body based on the characteristics of the radionuclide and path of exposure (inhalation or full body exposure) (issue: this is not plain English).
- Radiation exposure of the offsite population and the health effects caused by this exposure.

HOW ARE RADIOACTIVE MATERIALS MODELED TO MOVE DOWNWIND AND EXPOSE THE POPULATION?
Radioactive materials are released from plant buildings as aerosol particles in a continuous plume of steam and other gases. MACCS uses site-specific weather data to predict the downwind concentration of radioactive material in the plume and the resulting population exposures and health effects. MACCS then applies a statistical model to predict the average individual risk as a result of the variability in the weather.

SOARCA modeled people to be exposed to radiation by inhaling the aerosol particles and by direct shine from aerosol particles that have settled on the ground. Some 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. Some of this exposure occurs in the long term, after land is decontaminated and people are allowed to return home. SOARCA modeled people returning home based on guidance on when it would be safe to return home. For the Surry model, SOARCA uses habitability criteria from the U.S. Environmental Protection Agency "Manual of Protective Action Guides for Nuclear Incidents" to determine when the population can return to an area. For the Peach Bottom model, SOARCA uses Pennsylvania-specific habitability 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 to be exposed by eating food on which aerosol particles may have settled, because contaminated food will likely be banned.

<u>(Figure 6.1 is too complicated and should be deleted. Its level of detail is inconsistent with the rest of the brochure.)</u>

Figure 6.1 Information Used to Model Health Effects
[you can see colors and formatting on page 38 of PDF Version 2.5.5]



GRAPHICS AND SIDE BOXES FOR CHAPTER 6

WHAT INFORMATION IS IN THE MODELING?
How Is Radiation Measured?
Units that measure how much radioactive material decays over a period of time:
Curie (Ci)
Becquerel (Bq)

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

More information about radiation and radiation protection is at http://www.nrc.gov/about-nrc/radiation.html.

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

What Are NRC Regulations?
Dose Limits for Emergency Responders
Radiation dose limits for emergency responders to a nuclear accident:
5 rem— Any activity
10 rem— Protecting valuable property
25 rem— Lifesaving or protecting large population

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

Historical Perspective: What about the Thyroid Cancer after Chernobyl?
After Chernobyl, thousands of children were exposed to radioactive iodine which, over the years, resulted in an elevated incidence of thyroid cancer. These children became exposed to radioactive iodine mainly by eating contaminated foods. Although, tragically, several have died, many have been effectively treated following diagnosis, so the death rate is low. The SOARCA project excludes the food pathway because the availability of food for U.S. population is different. In essence, SOARCA assumes that if a severe nuclear reactor accident contaminated an environment, the population would be instructed not to eat the food and would have access to enough food from other areas that they would not need to consume the contaminated food—hence averting radioiodine exposure through ingestion. SOARCA does model exposure to the thyroid from inhaled radioiodine (Issue: what likelihood is assumed in MACCS for the effectiveness of treating thyroid cancer?).

Chapter 7
Results and Conclusions

This chapter summarizes SOARCA's results and conclusions.

The mitigated scenarios demonstrate that reactor designs, operator actions, and regulations can prevent consequences to the public from a severe accident (issue: This is awkward. This uses different language than in other parts of the brochure.). Additionally, the SOARCA team ran scenarios that demonstrate the consequences if newly available equipment is not used as additional mitigating actions (issue: in some cases, ran scenarios without using old equipment. For example, in the ISLOCA, the operators were assumed to not cross-tie to high pressure injection in the unaffected unit.). The unmitigated scenario results~~, presented in this chapter,~~ (issue: mitigated scenario results also are presented in this chapter) demonstrate that even in these cases the public health consequences are very low.

WHAT ARE THE RESULTS OF THE MITIGATED SCENARIOS?
In the modeling for all but one of the SOARCA scenarios, the operators were able to prevent a release of radioactive material. (The mitigated Surry TISGTR had a release.)

Early Fatality Risk—As a result, the mitigated scenarios show essentially (issue: what is meant by "essentially?" The output from the MACCS calculation for this case was "0.00.") no risk of early fatalities from radiation exposure.

Long-Term Cancer Fatality Risk—As a result, the mitigated scenarios show result is essentially no risk (issue: What is meant by "essentially no risk?" How does this prediction compare with the predictions in Table 7.1?) of long-term cancer fatalities.

Which operator actions can mitigate accidents?
The operators mitigate the station blackout events ~~(~~and TISGTR~~)~~ ~~accidents~~ (issue: wordy) either by manually operating the reactors' steam-driven safety injection pumps or by using portable diesel-driven pumps that are stored on-site. The operators mitigate the interfacing systems loss of coolant accident by using normal plant equipment due to the long time predicted until core damage.

WHAT ARE THE RESULTS OF THE UNMITIGATED SCENARIOS?

Early Fatality Risk—The unmitigated scenarios result in essentially no risk of early fatalities. Even though these scenarios lead to core damage, the release of radioactive material occurs after long periods of time which allows for protective actions of the population (including evacuation, sheltering, and relocation). Therefore, in these situations, the public is not initially exposed to large amounts of radioactive material.

Long-Term Cancer Fatality Risk—In these scenarios, the individual risk of a long-term cancer fatality is calculated to be very small—regardless of which distance interval or low-dose calculation model is used. ~~Tables 7.1, 7.2, and 7.3~~ Table 7.1 summarizes the results based on the linear-no-threshold model for estimating the risk within 10 miles of each plant.

To gain perspective, it can be helpful to compare these results with the NRC Safety Goal and average risk of a cancer fatality in the United States[1]. The NRC's Safety Goal states that cancer fatality risk that might result from

[1] *We acknowledge that these comparisons have limitations. Relative to the safety goal comparison, SOARCA does not examine all scenarios typically considered in a probabilistic risk assessment, even though we did include the important scenarios (issue: I have not seen a PRA where the risks from all of the scenarios are summed and compared with the Safety Goal. For example, NUREG-1150 separately compares internal events scenarios with the Safety Goal and separately compares fire-initiated events with the Safety Goal.). On the other hand, SOARCA developed more realistic estimates of risk for the important scenarios. Relative to the U.S. average individual risk of a cancer fatality comparison, we acknowledge that the sources of an individual's cancer risk include a complex combination of age, genetics, lifestyle choices, and other environmental factors whereas the consequences from a severe accident at a nuclear plant are involuntary and unlikely to be experienced by most individuals. Despite these limitations, we believe that comparing risks of cancer fatality to the NRC safety goal and U.S. average individual risk of a cancer fatality provides context that may help the reader to understand the contribution that risks of nuclear power plant accidents make.*

nuclear power plant operation to the population near a nuclear power plant should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes. According to the American Cancer Society, cancer deaths in the United States were 180.7 per 100,000 in 2006. One-tenth of one percent of this number equals 1.8 deaths per 1 million people. As shown in Table 7.1, calculated cancer fatalities for the postulated events in this study range from 1 in 10 billion to 7 in 100 billion. This is significantly lower than the NRC Safety Goal and the average risk of a cancer fatality from all causes.

Table 7.1 Unmitigated Scenarios[2,3]

	Peach Bottom Long Term Station Blackout	Peach Bottom Short Term Station Blackout	Surry Long Term Station Blackout	Surry Short Term Station Blackout	Surry Interfacing-Systems Loss of Coolant Accident	Surry Thermally-Induced Steam Generator Tube Rupture
How likely is the accident to occur?	3 events in one million reactor years	3 events in 10 million reactor years	2 events in 100,000 reactor years	2 events in one million reactor years	~~8~~3 (issue: this number is incorrect)events in 100 million reactor years	4 events in 10 million reactor years
From the initiating event, how long until the release begins?	20 hours	8 hours	46 hours	26 hours	10 hours	3.5 hours
How much of the core inventory of radionuclides is released during the first ~~after~~ 48 hours? (issue: clarification)	Iodine - 4% Cesium - 2%	Iodine - 11% Cesium - 2%	Iodine – less than1% Cesium – less than1%	Iodine - 1% Cesium – less than1%	Iodine - 9% Cesium - 9%	Iodine – 1% Cesium - <1%
What is the risk —the annual average individual risk of a long-term cancer fatality for this scenario within 10 miles of the plant?	8 in 10 billion	7 in 100 billion	7 in 10 billion	1 in 10 billion	2 in 10 billion	1 in 10 billion

[2] *The full technical report, NUREG 1935, reports these results for an individual located within the following distance intervals: 0-20, 0-30, 0-40, 0-50 miles from the reactor. The health effects results for the threshold models. This information in chapter x*

[3] *The results reported in the table are based on the analyses in which newly available equipment is not used as additional mitigating actions.*

HOW DOES THE NRC DETERMINE THE VALIDITY OF THIS STUDY?

Peer Review— Peer reviews, in which experts not involved in a research project examine the results, help improve the work by identifying the project's strengths and weaknesses ~~of~~. The SOARCA team assembled a panel of external experts in the field of risk analysis and severe accident research. This group reviewed SOARCA's methodology, underlying assumptions, results, and conclusions to ensure that they are defensible and state-of-the-art. The SOARCA team has incorporated the experts' feedback, to date, into the document. When the peer review is complete, the feedback will be incorporated into the final published document.

Uncertainty— Scientific research strives for valid results based on high-quality data and reasonable assumptions. However, because data can be uncertain, researchers work systematically to identify any weaknesses in data and assumptions and consider alternatives. This step is an important part of making research transparent. NRC staff used sensitivity analyses to compare the how varying individual input assumptions affect the outcomes. Additionally, the NRC plans to use a statistical approach to assess the uncertainties in a more integrated fashion.

WHAT DO SOARCA RESULTS INDICATE ABOUT CONSEQUENCES OF SEVERE ACCIDENTS?
The results of this project indicate that nuclear power plants' robust design and their defense-in-depth engineering prevent accidents and protect the public should an accident occur (issue: overstatement). More than 40 years (issue: why 40 years?) of improvements in plant designs, operations, mitigation measures, and emergency preparedness have continued to increase overall nuclear power plant safety. Other SOARCA insights for the analyzed scenarios include the following:

- Accidents can reasonably be mitigated. Newly incorporated mitigative measures, as implemented according to NRC rules, further protect the public from radioactive releases.
- For the unmitigated cases, a~~A~~ccident progression is several hours slower than previously thought, resulting in a delayed release of radioactive material.
- ~~Newly incorporated mitigative measures, as implemented according to NRC rules, further protect the public from radioactive releases.~~
- For the unmitigated cases, t~~T~~he modeled radioactive releases are delayed and relatively small.
- For the unmitigated cases, a~~A~~s a result, early fatalities from severe accidents at nuclear power plants are essentially zero and long-term cancer fatality risks are small.
- Results, while specific to the two nuclear plants selected, may be generally applicable for plants with similar designs. However, results may vary depending on individual plant designs and capabilities and effectiveness of emergency response for surrounding populations.

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WHAT DOES SOARCA HAVE TO DO WITH THE FUTURE OF COMMERCIAL NUCLEAR POWER?
The NRC began the SOARCA project prior to the commercial nuclear power industry's renewed interest in licensing new nuclear power plants. Regardless of this renewed interest, the NRC's mission is to regulate civilian uses of radioactive material (including commercial nuclear power plants) "to protect public health and safety, promote the common defense and security, and protect the environment." The NRC neither promotes nor opposes nuclear power plants; the NRC must ensure that if nuclear power plants operate, they do so safely. Under this safety framework, the NRC conducted SOARCA to better understand the impact of decades (issue: why decades?) of improved research, operations, and regulation on the consequences of hypothesized accident scenarios. SOARCA's results indicate that industry efforts to improve their plant designs and operations, as well as NRC developments in rigorous inspection methods, operator training, and emergency preparedness have improved nuclear power plant safety. The staff at the NRC will continue to diligently perform its responsibilities to protect public health and safety (issue: doesn't this go without saying?).

GRAPHICS AND SIDE BOXES FOR CHAPTER 7

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

As a resident of the U.S., how am I exposed to radiation?
SOARCA studies health effects in situations that a severe accident releases radiation to the public. To provide some perspective, people generally receive a 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 man-made sources account for the other 50 percent.
[the pie graph appears on page 45 of PDF Version 2.5.5]

Openness
As a Federal agency committed to serving the public, the NRC operates transparently and with respect for differing views of its stakeholders including the public. The results and methods of this research project are of great interest to many people. Therefore, the SOARCA team worked diligently to make the research methods of this project transparent and the results comprehensible in the publicly available technical reports (NUREG-1935, NUREG/CR-7008, and NUREG-CR-7009).

GLOSSARY

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 production and utilization facilities (issue: "production and utilization" is not plain English)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, which 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. Such enclosures are usually cylindrical with a dome-shaped top and made of steel-reinforced concrete.

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

Core Damage—Events leading to heat up of the reactor core to the point at which fuel damage is anticipated or uncovery and heat up 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 be damaged.

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

Emergency Planning Zone (EPZ)—The EPZ around each nuclear power plant helps 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.

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.

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

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

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 that are designed 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 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 alloy (Zircaloy) tubes to form fuel rods. Fuel rods are assembled into bundles called fuel assemblies which are loaded into the reactor core.

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

Severe Accident—A type of accident that may challenge safety systems at a level much higher than expected in the design basis.

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

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NUREG/CR-7009: *State-of-the-Art Reactor Consequence Analyses - MACCS2 Best Modeling Practices.*

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Backgrounder on Chernobyl Nuclear Power Plant Accident http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/chernobyl-bg.html
Backgrounder on Three Mile Island Accident http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/3mile-isle.html
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