ATTACHMENT 43

NUREG/BR-0359, "Modeling Potential Reactor Accident Consequences," (Jan. 2012) (Attach. 43)

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

PERSONAL STATEMENT FROM BRIAN SHERON

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The U.S. Nuclear Regulatory Commission's (NRC's) Office of Nuclear Regulatory Research is a congressionally mandated office that plans, recommends, and implements a program of nuclear regulatory research, standards development, and resolution of generic issues for nuclear power plants and other NRC-regulated facilities. We partner with other NRC offices, Federal agencies, industry research organizations, and international organizations to conduct these activities.

We started the State-of-the-Art Reactor Consequence Analyses (SOARCA) research project to estimate the potential health effects from the unlikely event of a commercial nuclear power plant accident releasing significant quantities of radioactive material into the environment. This project modeled a set of important accident scenarios for two plants, Peach Bottom and Surry, which represent two of the most common types of plants licensed in the United States. SOARCA considers plant design and operational changes not reflected in earlier assessments. The project also takes into account NRC's development of rigorous oversight processes and use of operating experience along with improvements in operator training and emergency preparedness. We've also incorporated decades of national and international research into the tools that NRC used to perform this study.

One of SOARCA's objectives is explaining severe-accident-related aspects of nuclear safety to NRC stakeholders. Stakeholders include members of the public along with Federal, State, and local authorities and the companies that hold NRC licenses to operate nuclear power plants. SOARCA meets this communication objective by documenting its results in reports: 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." Because the NUREG reports rely on highly technical explanations, this brochure was developed as a plain-language summary of SOARCA's methods, results, and conclusions. We invite you to read this brochure to understand how we used state-of-the-art methods to model these unlikely nuclear power plant accidents to understand their potential impact on public health and safety.

ACKNOWLEDGEMENTS:

This brochure was the result of the efforts of many NRC staff and the staff of its contractors, Sandia National Laboratories, and dycoda, LLC. The contributions of Dorothy Andreas (née Collins) and Scott Burnell are especially appreciated. The SOARCA team also wishes to thank Eric Knowles and Woody Machalek for their help with the graphics and layout.

The SOARCA project team consisted of the following individuals. This group also benefited greatly from the contributions of other selected NRC staff members, including senior risk analysts from NRC's headquarters and regional offices, in specific areas of the project.

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KEY RESULTS:

- When operators are successful in using available 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 much 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
MACCS2	MELCOR Accident Consequence Code System, Version 2
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
TISGTR	thermally induced steam generator tube rupture

PROJECT OVERVIEW

CHAPTER 1

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

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.

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 very detailed computerized plant models.

NRC, the nuclear power industry, 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 "10 CFR 50.54(hh) mitigation" enhancements are named after the relevant section of the NRC's regulations.
- 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.

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.

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 researched 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.

HOW ARE SEVERE ACCIDENTS AND POTENTIAL HEALTH 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, evacuation time estimates, and population.

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 a Severe Accident?

A severe accident is a type of accident that may challenge safety systems at a level much higher than expected.

A reactor accident occurs when the plant cooling water systems are no longer removing heat from the reactor fuel (the "core" of the reactor). 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, however, 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.

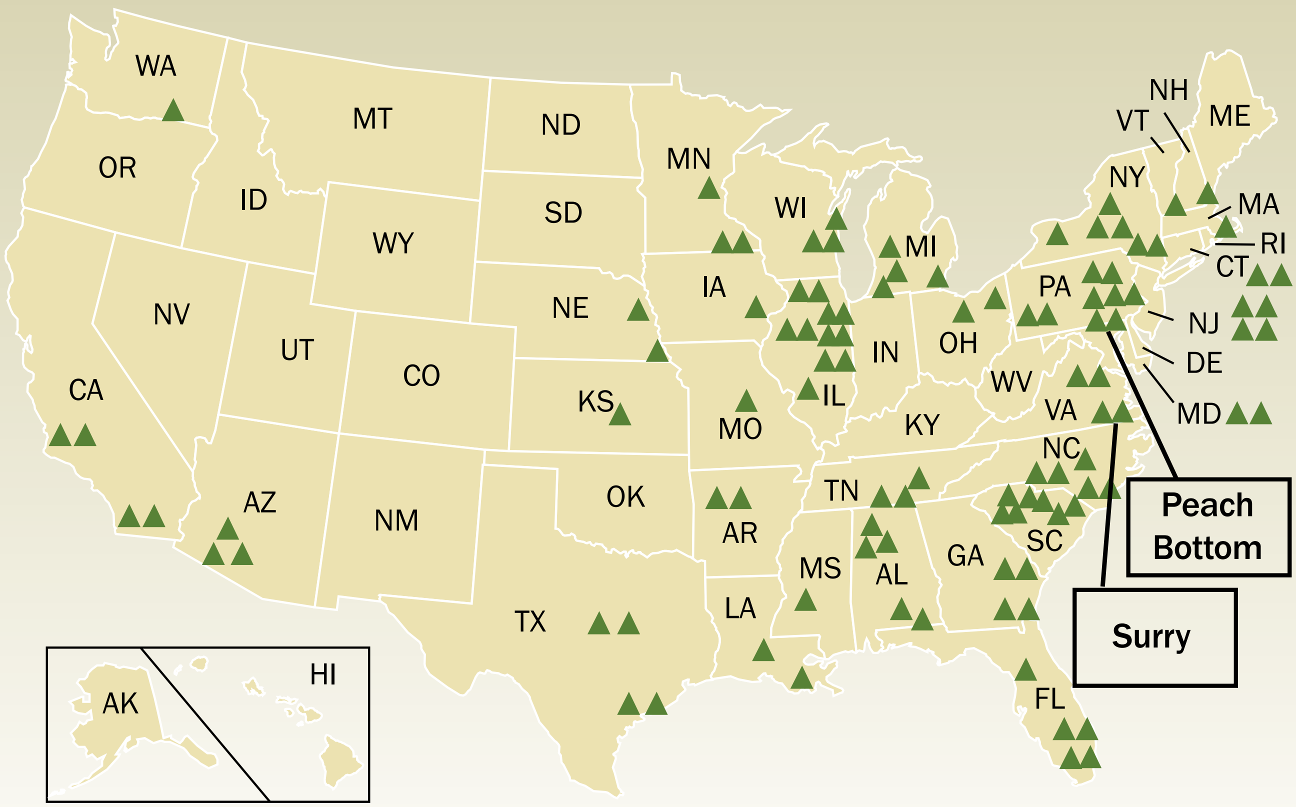
What Are NRC Regulations?

NRC works diligently to ensure safe operation of nuclear power plants, supporting safety by developing 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 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/.

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. The project team first decided it could learn more by rigorously and realistically analyzing a relatively small number of important accident scenarios, rather than carrying out less-detailed modeling of many 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 10 CFR 50.54(hh) mitigation measures (put in place after 9-11 for responding to fires and explosions) in other accident scenarios. 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 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.

Figure 1.1 Locations of All Operating Reactors in the United States.



Licensed to Operate (104)

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. 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 (a standing group of nuclear safety experts) also reviewed the project and provided comments. The SOARCA team has incorporated the experts' feedback into the reports.

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 is taking a systematic look at potential sources of uncertainty and their impact on SOARCA results in a separate uncertainty analysis. The uncertainty analysis uses a statistical approach to assess the uncertainties in a more integrated fashion.

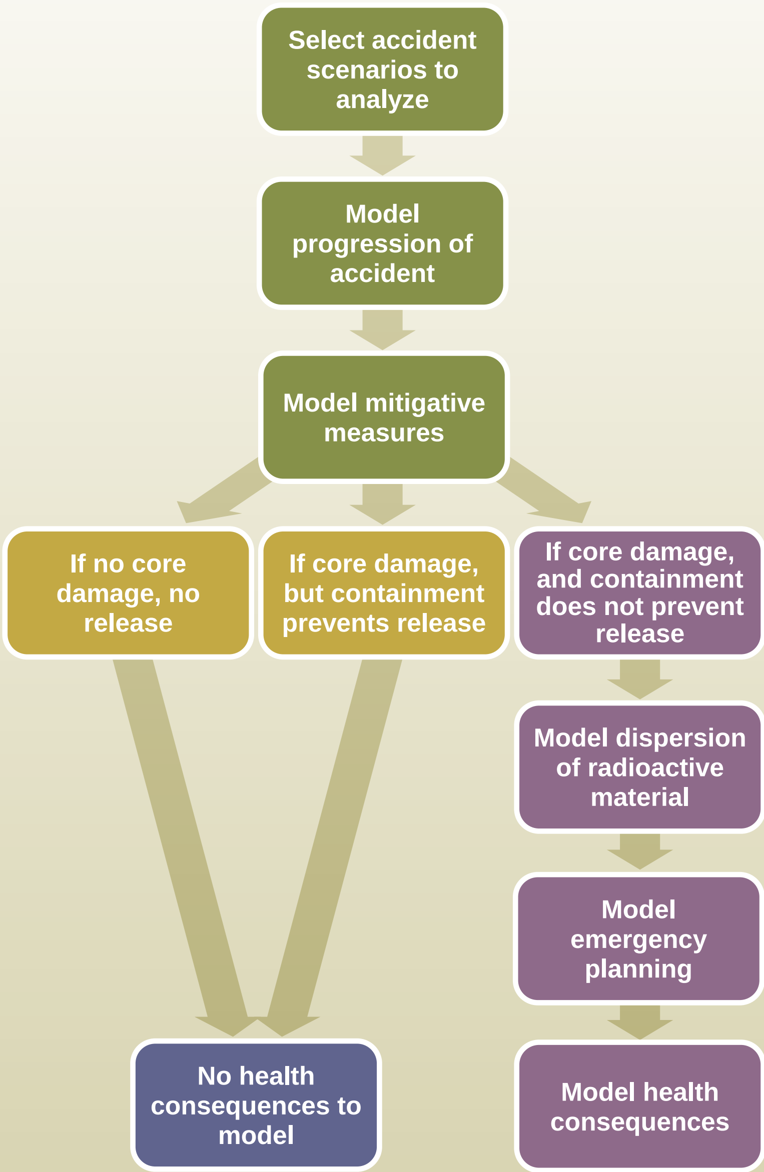
What Computer Codes Were Used for SOARCA?

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, MACCS2 (MELCOR Accident Consequence Code System, Version 2), 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.

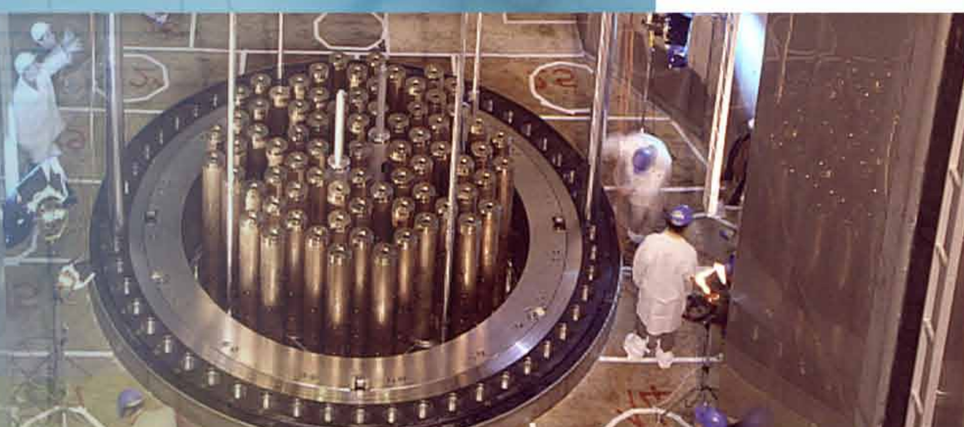
An expert panel review of the MACCS2 code and SOARCA's MACCS2 modeling choices was conducted in August 2006, prior to the start of specific work as part of the Peach Bottom and Surry analyses. This expert panel review and the NRC staff recommendations influenced much of the development that has been 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.

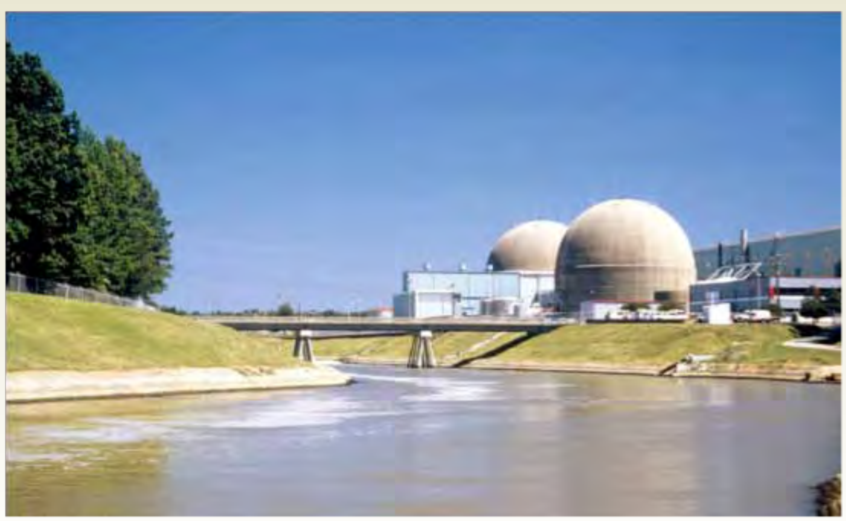
WHICH PLANTS DID SOARCA STUDY?

SOARCA analyzed an example of each major type of operating U.S. nuclear reactor: a boiling-water reactor (BWR) and a pressurized-water reactor (PWR). The project team solicited volunteers from the nuclear industry to participate in the project. The Peach Bottom Atomic Power Station in Pennsylvania and the Surry Power Station in Virginia volunteered and are the focus of this report. Peach Bottom is a General Electric-designed BWR with a Mark I containment. Surry is a Westinghouse-designed PWR with a large dry containment. These two plants, depicted in Figure 2.1, also were part of earlier studies.

WHAT ARE THE DIFFERENCES BETWEEN REACTOR TYPES?

Figures 2.2 and 2.3 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 2.1 Peach Bottom Atomic Power Station (top) and Surry Power Station (bottom).



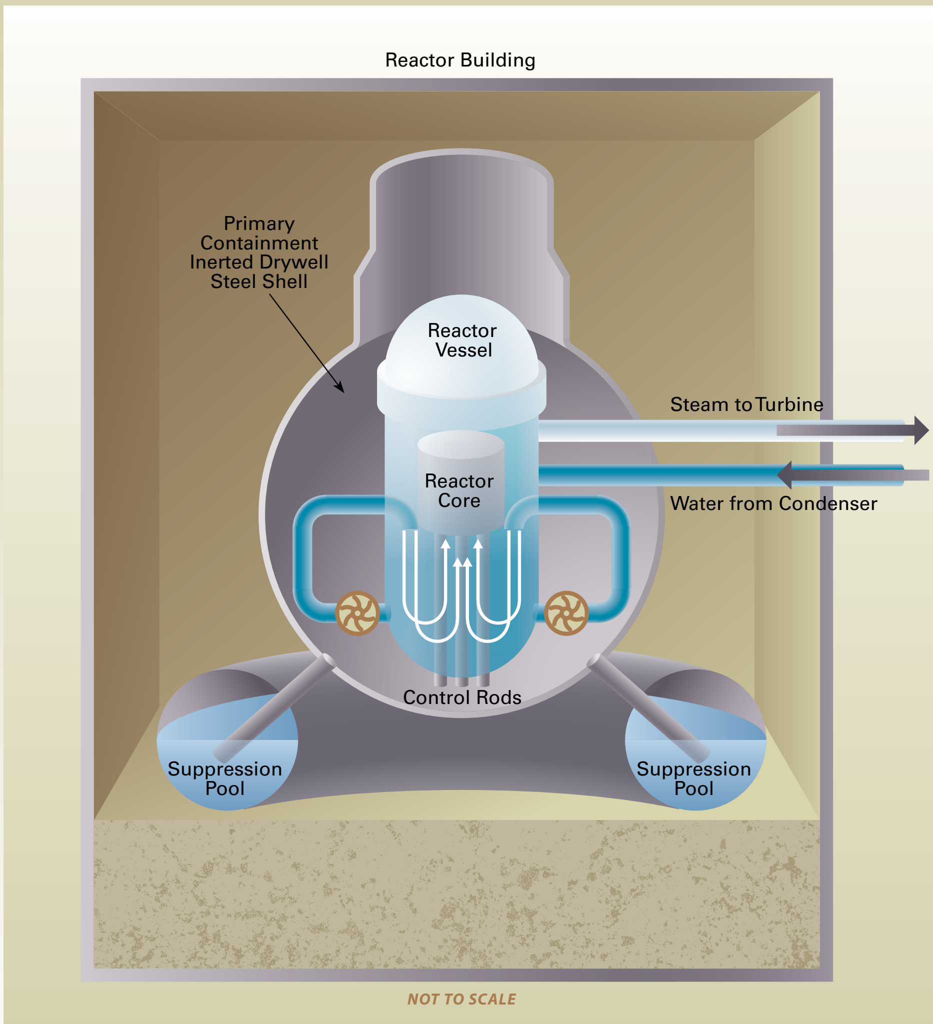
HOW WERE SCENARIOS SELECTED?

The project team sought to focus its attention and resources on the important severe accident scenarios for Peach Bottom and Surry 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 external events information from 2005. 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 CDFs to identify the most risk-significant groups. SOARCA analyzed scenarios with a CDF equal to or greater than 1 in a million reactor-years. SOARCA also

Figure 2.2 Typical U.S. Boiling-Water Reactor.

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.

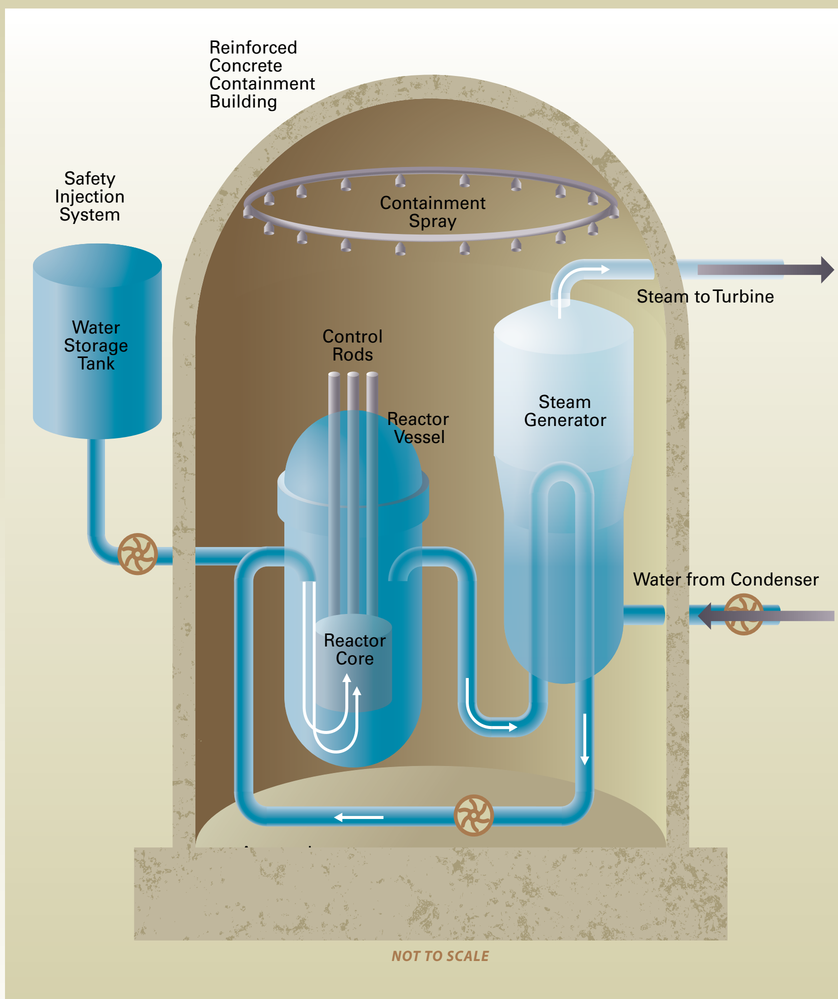
sought to analyze scenarios leading to an early failure or bypass of the containment where the CDF is equal to or greater than 1 in 10 million reactor-years, since these scenarios have a potential for higher consequences and risk. This approach allowed a more detailed analysis of accident consequences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

WHAT ACCIDENT SCENARIOS WERE ANALYZED?

For both Peach Bottom and Surry, 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; however, SOARCA modeled the earthquake scenario

Figure 2.3 Typical U.S. Pressurized-Water Reactor.

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 saturated 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.



because this presented the most severe challenge to the plant operators as well as offsite emergency responders, and had the highest probability of occurring.

Long-Term Station Blackout (LTSBO)—In this scenario, the plant loses 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 site loses all power (even the batteries), all of the safety systems immediately become inoperable, and core damage occurs in the "short term."[1] The STSBO scenario starts with a more extreme earthquake than the one that starts the LTSBO.

In addition, the team analyzed two scenarios for Surry in which radioactive material could potentially reach the environment by bypassing containment features. These are discussed in more detail in Chapter 4.

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.

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 and Surry both 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.

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 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 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:

[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".*

- 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 extremely high temperatures.
- 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 (MACCS2) 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. 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 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	Probability/Frequency
Surry	Long-Term Station Blackout	1 event in ~ 50,000 years
Peach Bottom	Long-Term Station Blackout	1 event in ~ 300,000 years
Surry	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: 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. Due to lack of ac power to pump water into Units 1 through 3 to cool the nuclear fuel, some of the fuel melted. 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, and Japan continues work on stabilizing these plants.

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.

An appendix to the main SOARCA report briefly compares and contrasts what we currently know about Fukushima with insights from the Peach Bottom SOARCA analyses. The NRC Web site has additional information on the Fukushima accident and NRC's response:

- http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/fs-japan-events.html
- http://www.nrc.gov/japan/japan-info.html

Comparison of Fukushima Accident to SOARCA Analyses

The SOARCA study had nearly completed its peer review when the Fukushima Dai-ichi accident occurred on March 11, 2011. Following the accident, 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.

Based on limited information currently available, the Fukushima accident is in some ways similar to a few of the Peach Bottom scenarios analyzed in SOARCA. 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 was 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.

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 and radioactive material into the containment building. A very small amount of radioactive material reached the environment. 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.

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 designing and operating 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.

Security-related (10 CFR 50.54(hh)) 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 fire or explosions.

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 security-related 10 CFR 50.54(hh) mitigating actions in responding to potential accident conditions. This equipment and procedures were intended to be used to maintain or restore safety functions under circumstances associated with loss of large areas of the plant due to explosions or fire. The NRC anticipates, however, that 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 and Surry 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.

Unmitigated Case—To understand the value of 10 CFR 50.54(hh) 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) measures and certain other actions that would prevent core damage. 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. These actions respond to design-basis events too, for which operators have more specific procedures and frequent training. The unmitigated cases modeled the sequence of events that 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. Figure 3.1 compares the mitigated and unmitigated timelines for the Peach Bottom long-term station blackout scenario from the blackout until the release starts (for the unmitigated case).

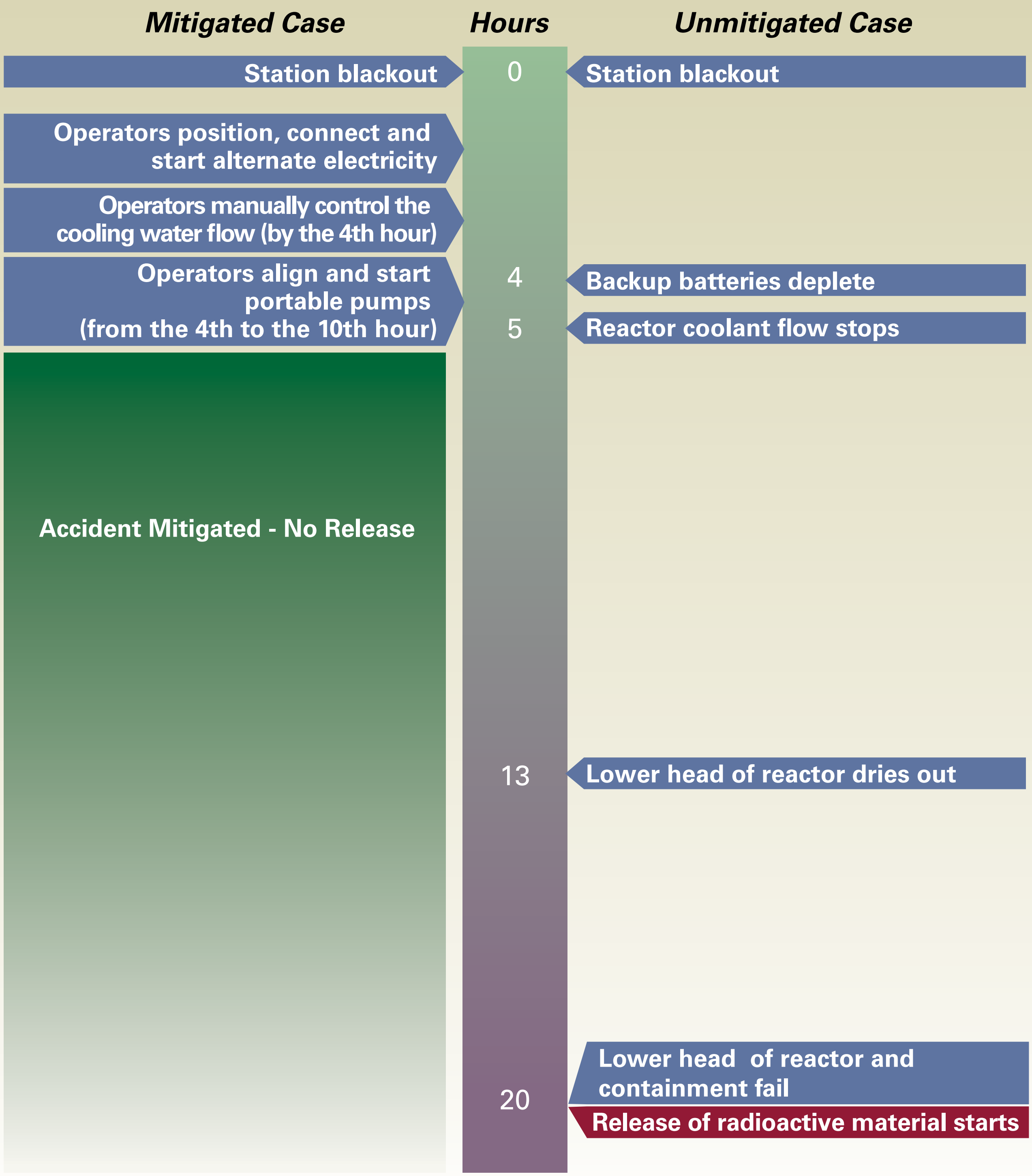
Historical Perspective: How Have Mitigation Capabilities Improved Since 9/11?

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). These mitigation measures include the following for the two plants analyzed in SOARCA:

- 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.



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?

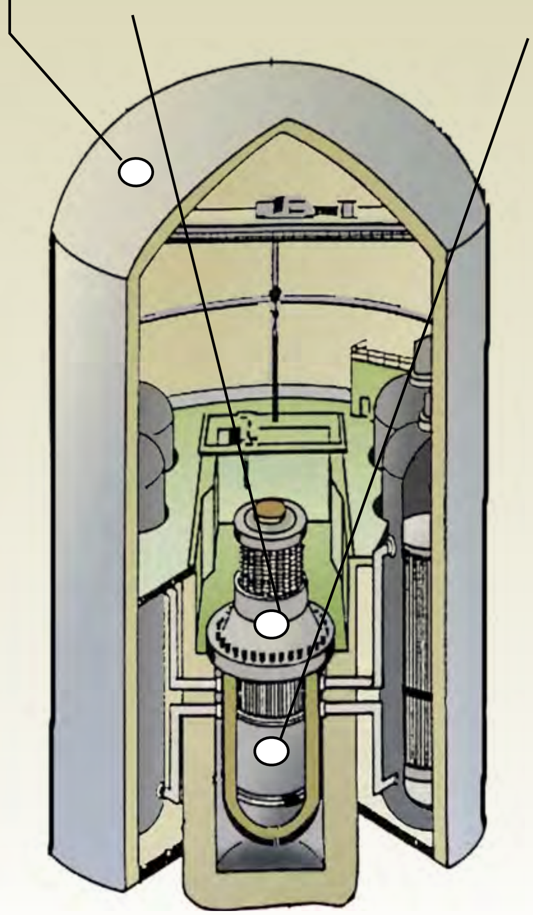
SOARCA took a detailed approach to considering radioactive substances, or radionuclides. In SOARCA, MELCOR calculations of reactor accident response are based on realistic estimates of decay heat generated by the radionuclides in the reactor core. MELCOR organizes the radionuclides by chemical similarity 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. The offsite consequences computer code (MACCS2) tracks radionuclides based on how long they remain radioactive, their biological importance, and how much is expected to be released from the core.

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 and the 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. Surry has a large dry, containment where the interior pressure is kept lower than atmospheric pressure. Surry's containment is expected to perform similarly during a severe reactor accident.

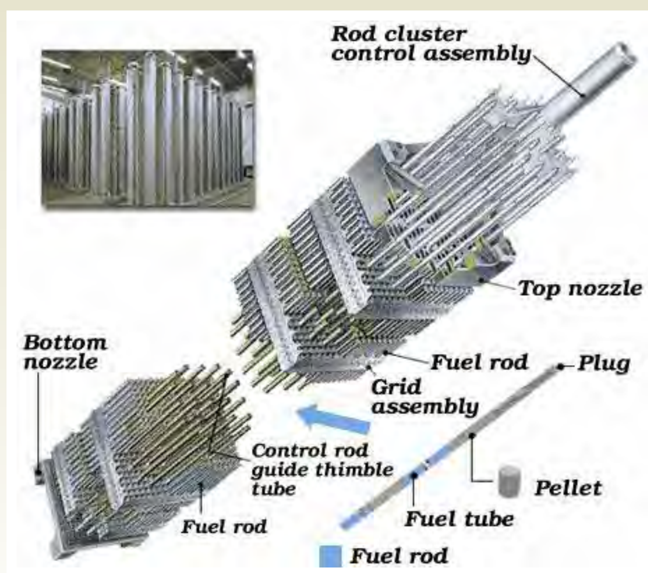


Diagram of components of a reactor fuel assembly

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.

Other radionuclides—MELCOR and MACCS2 also consider other radiological inventory in the analysis, and consequence results in NUREG-1935 include health effects from the 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 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 during the first 48 hours of the accident.

Peach Bottom Scenarios (Unmitigated Cases)

Long-Term Station Blackout— 20 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.

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.

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)

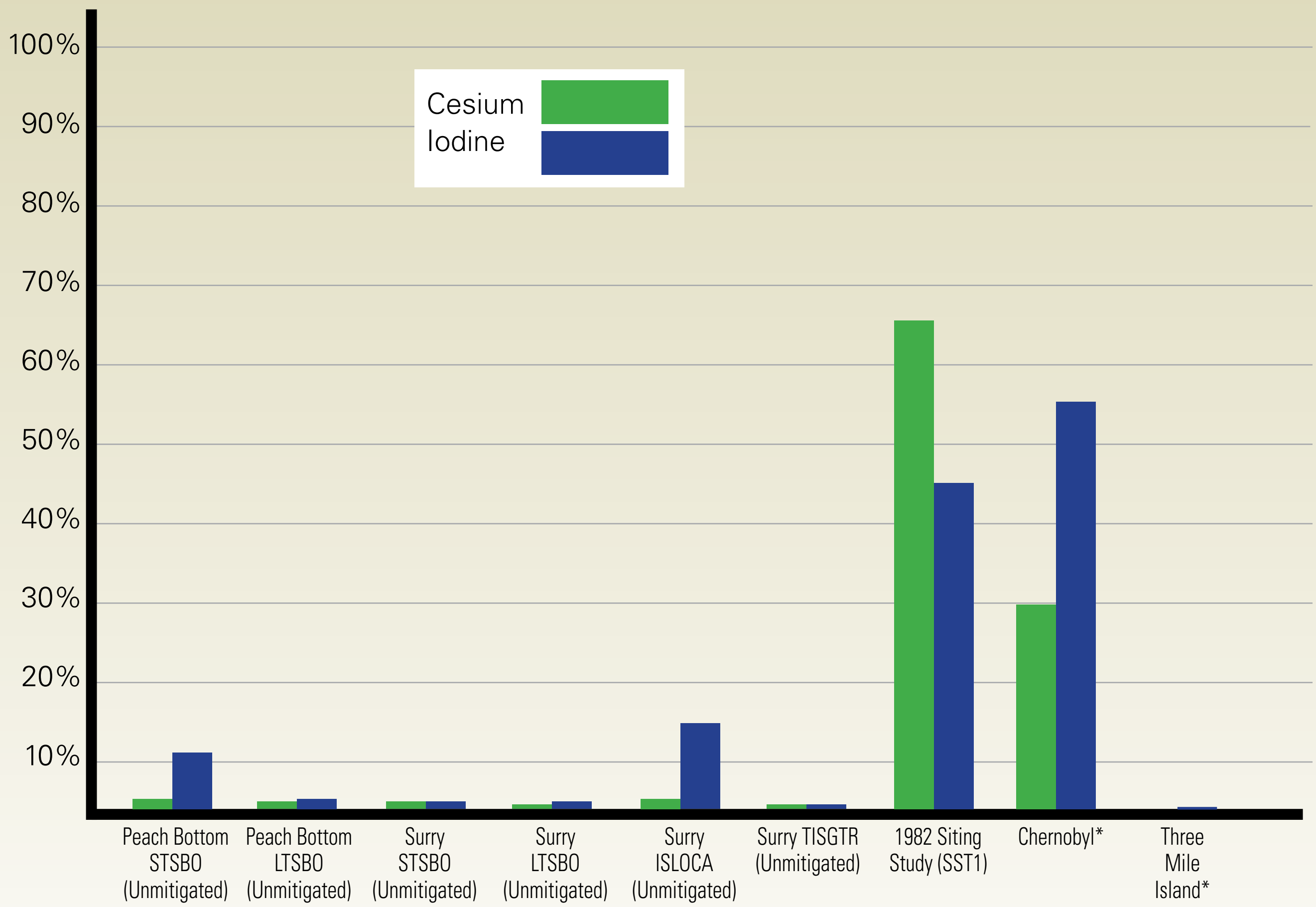
Long-Term Station Blackout— About 45 hours after the scenario begins, the pressure in the containment building exceeds the building's limits, tearing the containment liner and cracking the reinforced concrete.

Short-Term Station Blackout— About 25 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.

For the two analyzed Surry station blackout events, while the fuel is overheating, radioactive material enters the containment building through ruptured reactor coolant system piping. 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.

Figure 4.1 Percentages of Iodine and Cesium Released to the Environment During the First 48 Hours of the Accident for SOARCA Unmitigated Scenarios, 1982 Siting Study (SST1), and Historical Accidents.

This figure compares how much iodine-131 and cesium-137 that are normally in the reactor core gets released in each accident scenario. The SOARCA unmitigated releases are much smaller than estimated in the earlier 1982 Siting Study Siting Source Term 1 (SST1) case. Also note that these releases can begin as early as 3.5 hours (for Surry STSBO with TISGTR) to as late as 45 hours (for Surry LTSBO), and some of these releases develop over a period of time. For comparison, releases from the Chernobyl and Three Mile Island accidents are included.



* 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.

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

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 during the first 48 hours?**	
	Mitigated Case	Unmitigated	Mitigated Case	Unmitigated
Peach Bottom Long-Term Station Blackout	no release	20 hours	no release	Iodine: 2% Cesium: <1%
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: <1% Cesium: <1%
Surry Short-Term Station Blackout	no release modeled in MACCS2	25 hours	no release within 48 hours*	Iodine: 1% Cesium: <1%
Surry Thermally Induced Steam Generator Tube Rupture	3.5 hours	3.5 hours	Iodine: <1% Cesium: <1%	Iodine: 1% Cesium: <1%
Surry Interfacing Systems Loss–of-Coolant Accident	no release	13 hours	no release	Iodine: 16% Cesium: 2%

* 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.

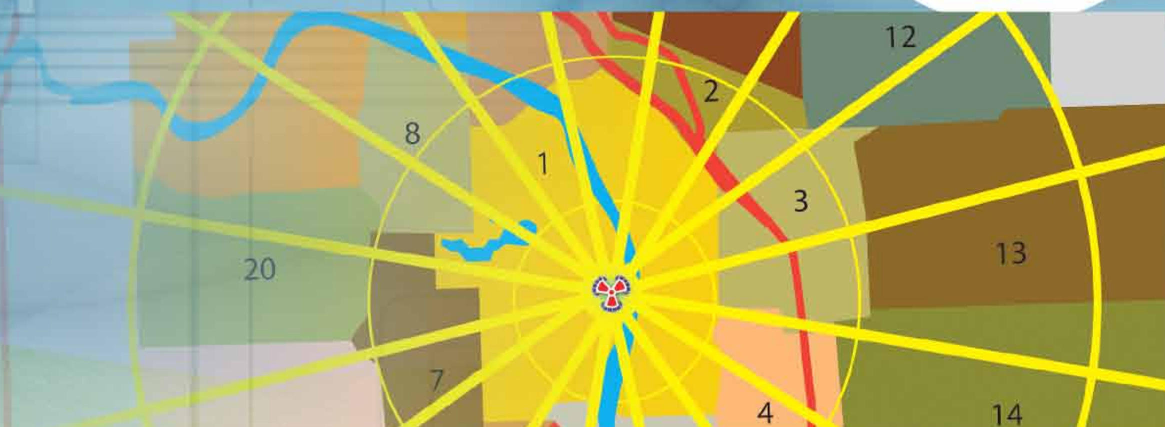
As supported by the SOARCA analyses, it is shown that the accidents evaluated could be mitigated through the actions of the onsite and offsite response agencies. The evaluation of the mitigation of source term and truncation of the accident at 48 hours further expands upon the response resources through identification of corporate, local, State, and Federal offsite resources. The responsibilities and resources of each of these organizations are described in onsite and offsite emergency response plans. These response organizations would mobilize upon request and as needed to respond to a severe nuclear power plant accident. These resources are in addition to the mitigative actions by the licensee though the use of safety and security enhancements, including SAMGs and 10 CFR 50.54(hh) mitigation measures. At Fukushima Dai-ichi, power supplies arrived onsite in less than 12 hours (INPO 11-005).

pipe surfaces and safeguards building (next to containment) filters, with a fraction of it entering the environment.

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.

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. In all scenarios except one, the releases' delayed timing (even without mitigative actions) allowed time to evacuate the local population. In that one scenario, evacuation began at the time of radiation release. However, the calculated individual long-term cancer fatality risk within 10 miles of the plant is similar to the other scenarios analyzed because of the lower probability of core damage due to this scenario.

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.

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 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[2].
- 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 some of the groups. Other population groups modeled include people who leave on their own initiative prior to the evacuation order as well as 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 5.1 illustrates the modeled timing of the unmitigated Peach Bottom LTSBO scenario and the timing of emergency response. Because the analyzed accident scenarios take several hours to start releasing radioactive material to the environment, this 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).

2 2010 U.S. Census data was not used because most calculations were already completed by the time it was released. Changes in population over the last decade are not expected to have a significant impact on any of the reported individual cancer fatality risks.

What Are NRC Regulations?

Emergency Plans

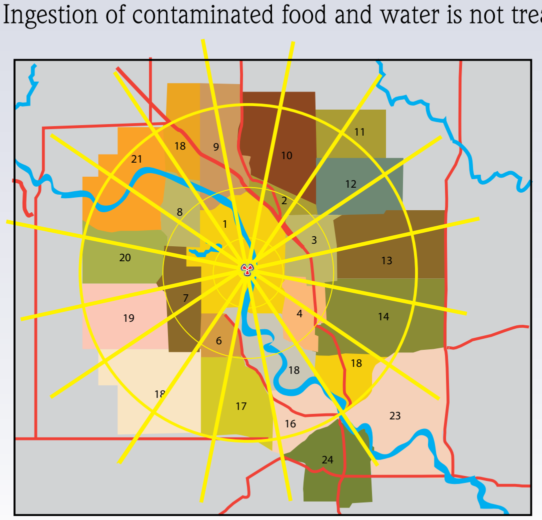
The planning standards of 10 CFR 50.47, "Emergency Plans," require nuclear plant licensees to develop comprehensive emergency response plans that include the support of State and local response organizations. Licensees must establish procedures to immediately notify offsite authorities of an emergency 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.

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.

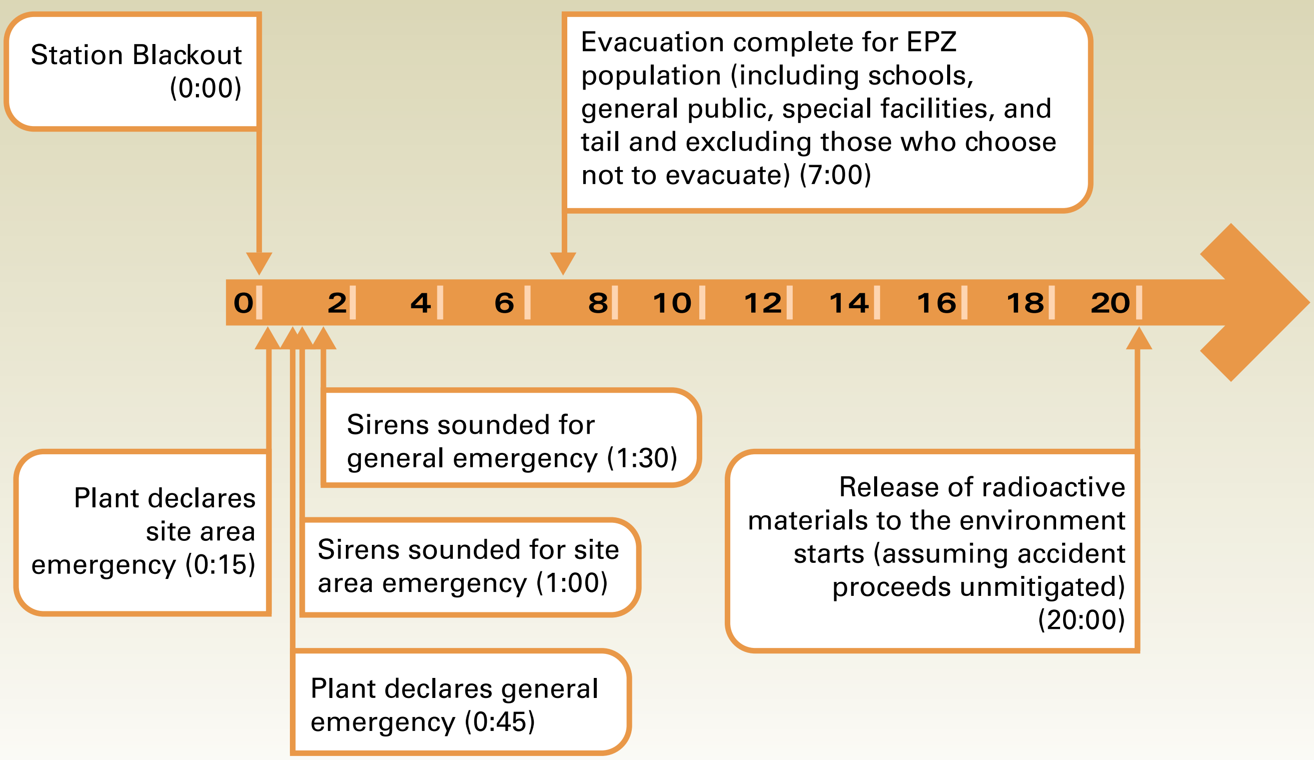


NRC Staff during an emergency preparedness drill

However, the MACCS2 modeling showed that seismic impacts for these two sites did not impact risk calculations because seismic impacts only affect the immediate phase of the accident when people are 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.

Figure 5.1 shows that groups are sheltered and evacuated before radioactive release begins. 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 Evacuation Timing for Peach Bottom Unmitigated Long-Term Station Blackout (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 fatalities. 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.*

Truncation 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 truncation or cutoff doses below which the cancer risk is not quantified. When comparing offsite consequence results for the linear-no-threshold model and truncation model, these truncation values make the already small long-term cancer fatality risk values even smaller (by orders of magnitude in some cases).

SOARCA uses two dose truncation values:

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

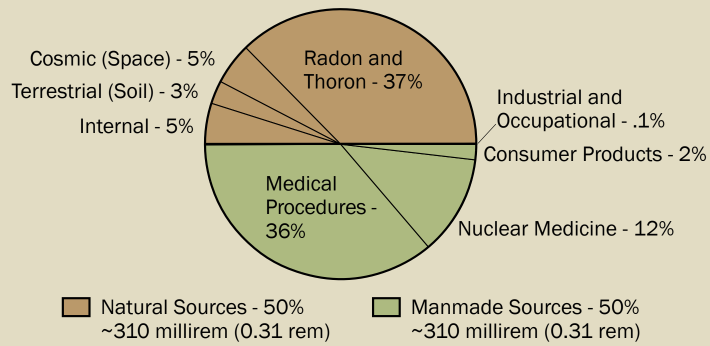
5 rem (5,000 mrem) per year with a 10 rem lifetime cap—This value was chosen based on the Health Physics Society position statement in "Radiation Risk in Perspective" (July 2010).

** 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.*

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.

The MACCS2 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.

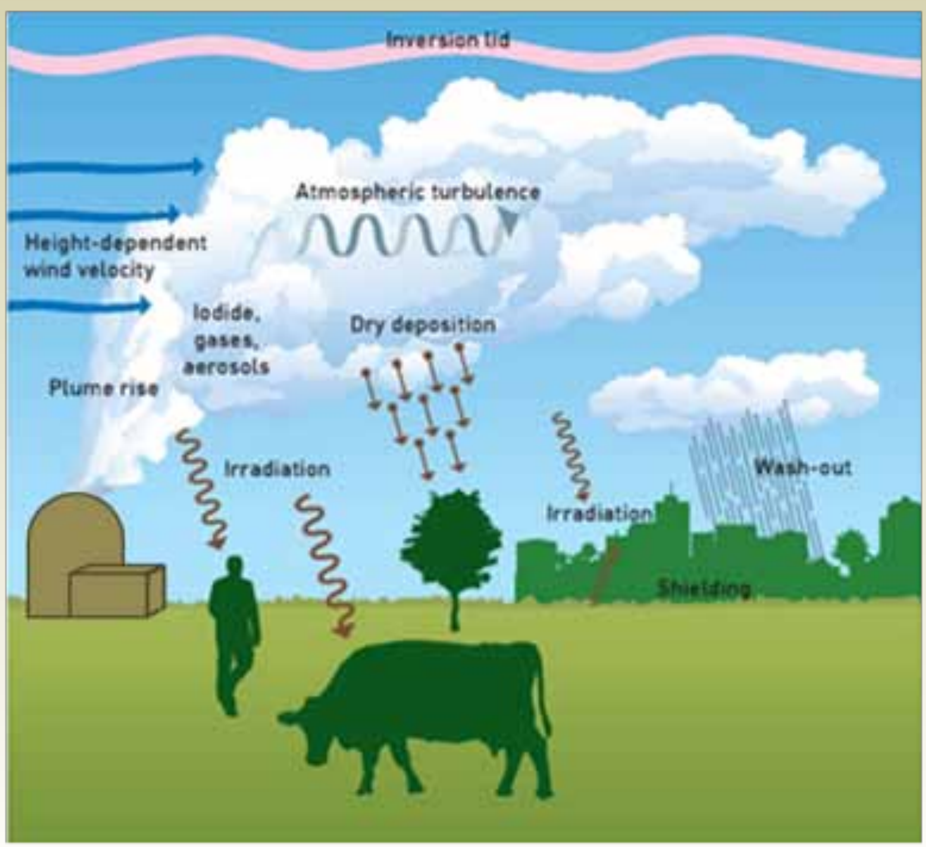
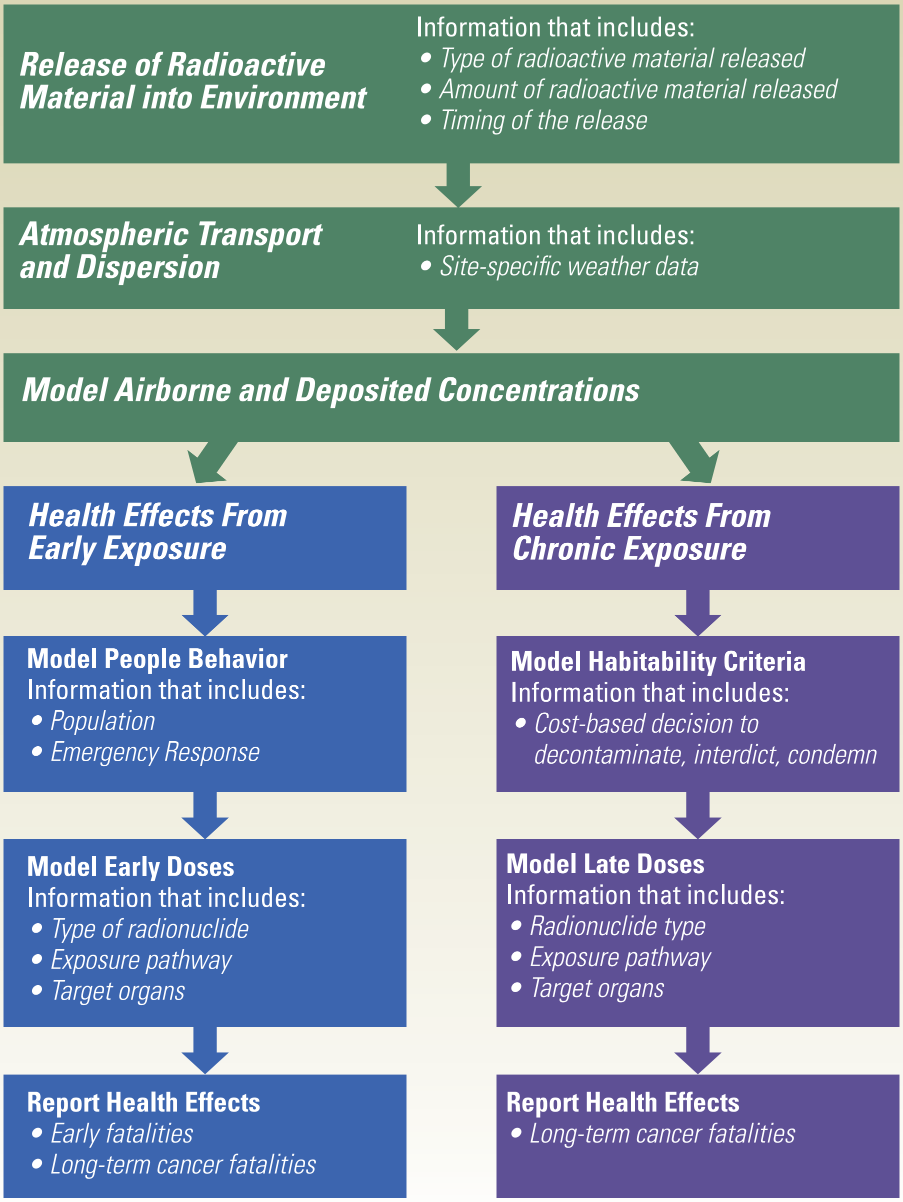


Figure 6.2 Information Used to Model Health Effects in SOARCA



WHAT INFORMATION IS INCLUDED IN THE MACCS2 MODELING?

- 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. MACCS2 uses site-specific weather data to calculate the downwind concentration of radioactive material in the plume and the resulting population exposures and health effects. MACCS2 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 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 emergency plans will prevent distribution of contaminated food and because of the expected availability of uncontaminated food from other areas.

RESULTS AND CONCLUSIONS

CHAPTER 7

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.

WHAT ARE THE RESULTS OF THE MITIGATED SCENARIOS?

All mitigated cases of 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.

Early Fatality Risk—As a result, the mitigated scenarios show zero risk of early fatalities from radiation exposure.

Long-Term Cancer Fatality Risk—As a result, the mitigated scenarios result in either zero risk or very small risk of long-term cancer fatalities, depending on the specific scenario.

HOW WOULD OPERATOR ACTIONS MITIGATE ACCIDENTS?

The operators mitigate the station blackout and thermally induced steam generator tube rupture (TISGTR) accidents by manually controlling pumps to inject water to keep the reactor core covered and cooled. The reactor's steam-turbine-driven safety injection pumps would be used in conjunction with portable diesel generators (stored onsite) to provide instrument readings. The operators mitigate the interfacing systems loss-of-coolant accident (ISLOCA) scenario by using normal plant equipment and procedures to ensure sufficient flow of water through the reactor coolant system to keep the core cooled.

WHAT ARE THE RESULTS OF UNMITIGATED SCENARIOS?

Early Fatality Risk—The unmitigated scenarios result in essentially zero risk of early fatalities. Although these unmitigated scenarios result in core damage and release of radioactive material to the environment, the release is 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.

Long-Term Cancer Fatality Risk—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) or low-dose calculation model is used. Table 7.1 summarizes the results based on the linear-no-threshold dose response model for estimating the risk for individuals located within 10 miles of each plant.

Table 7.1 SOARCA Results: Mitigated and Unmitigated Cases

	About how likely is the accident to occur?	About what is the annual average risk* of a long-term cancer fatality for this scenario for an individual located within 10 miles of the plant?	
		Mitigated Case	Unmitigated Case
Peach Bottom LTSBO	1 event in 300,000 reactor years	zero	1 in 3 billion
Peach Bottom STSBO	1 event in 3 million reactor years	zero	1 in 20 billion
Surry LTSBO	1 event in 50,000 reactor years	zero	1 in 1 billion
Surry STSBO	1 event in 500,000 reactor years	zero**	1 in 10 billion
Surry TISGTR	1 event in 3 million reactor years	1 in 10 billion	1 in 10 billion
Surry ISLOCA	1 event in 30 million reactor years	zero	1 in 100 billion

* 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.

** 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.

WHAT DO SOARCA RESULTS INDICATE ABOUT CONSEQUENCES OF SEVERE ACCIDENTS?

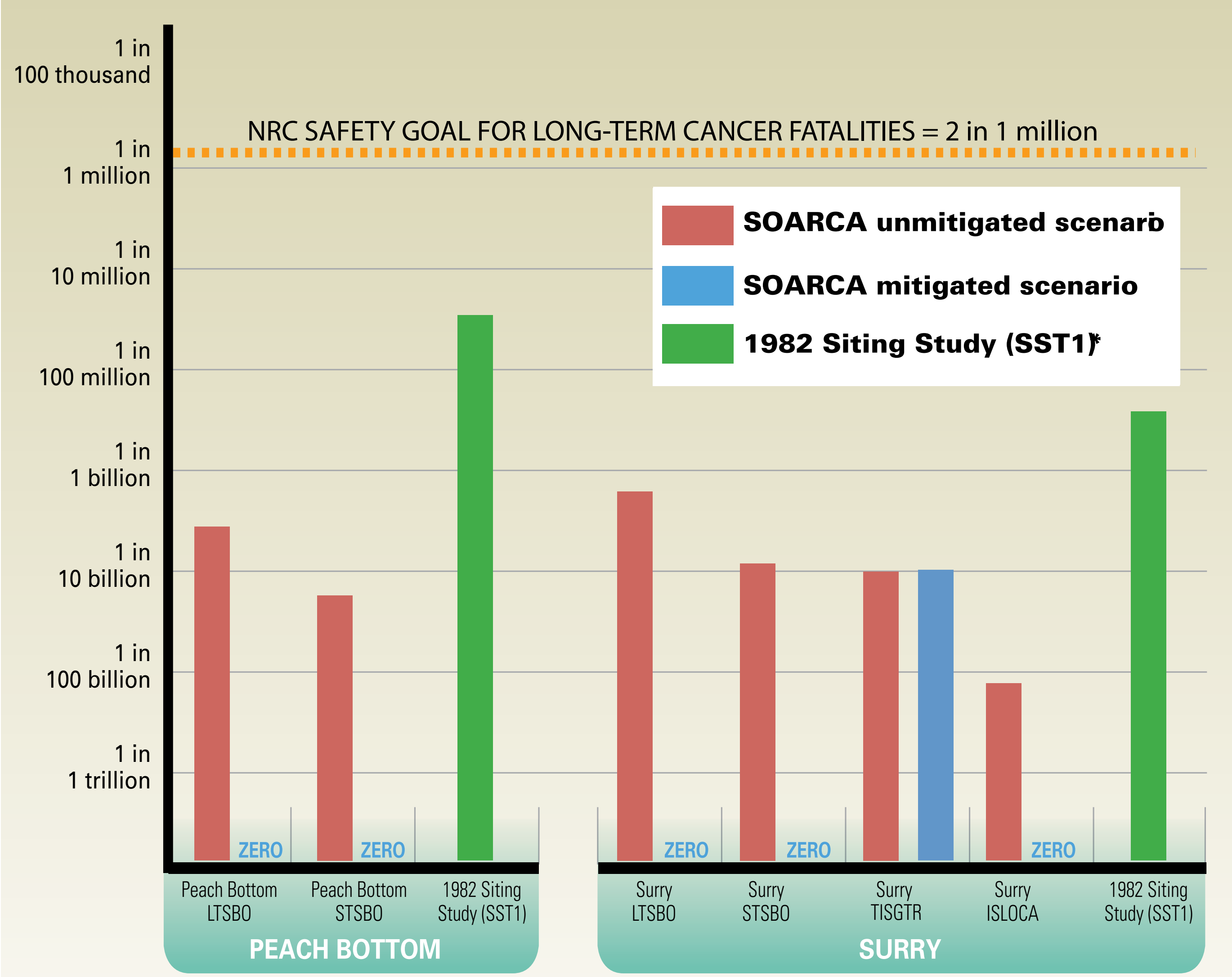
The SOARCA results for the two plants analyzed are as follows. These results, while specific to Peach Bottom and Surry, may be generally applicable to plants with similar designs. Additional work would be needed to confirm this, however, since differences exist in plant-specific designs, procedures,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

- 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 much 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 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 (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.

Figure 7.1 compares SOARCA's scenario-specific latent cancer fatality risks for an individual within 10 miles of the plant to the NRC Safety Goal and to an extrapolation of the 1982 Siting Study SST1 results.

Figure 7.1 Scenario-specific Risk of Dying From Long-Term Cancer for an Individual within 10 Miles of the Plant (per reactor-year).



* The 1982 Siting Study did not calculate risk of long-term cancer deaths. Therefore, to compare the 1982 Siting Study (SST1) results to SOARCA's results for risk of long-term cancer death, the SST1 release was put into the MACCS2 code files for Peach Bottom and Surry unmitigated STSBO calculations.

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. For example, the 1982 Siting Study calculated 92 early fatalities for Peach Bottom and 45 early fatalities for Surry for the siting source term 1(SST1) release of radioactive material. In contrast, SOARCA calculated essentially zero early fatalities for both 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. Figure 7.1 shows this comparison for individuals within 10 miles of the plant. 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 from returning home after the accident.

HOW DO SOARCA RESULTS COMPARE TO THE NRC SAFETY GOAL AND OVERALL U.S. CANCER RISKS?

To gain perspective, it can be helpful to compare the SOARCA results with the NRC Safety Goal and the average annual risk of a cancer fatality in the United States. The NRC's Safety Goal states that cancer fatality risk to the population near an operating nuclear power plant should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all causes. According to the American Cancer Society, 1 in 553 people died of cancer in the United States in 2006. One-tenth of one percent of this number equals about 1 death per 553,000 people. The NRC Safety Goal for long-term cancer fatality risk from nuclear power plant operation (i.e., 2 in 1 million) is set at 1,000 times lower than the sum of cancer fatality risks resulting from all other causes (i.e., ~ 2 in 1,000) per year. As shown in Table 7.1, SOARCA's calculated cancer fatality risks for the postulated events in this study range from about 1 in 1 billion per year to 1 in 100 billion per year.

The calculated cancer fatality risks from the scenarios analyzed in SOARCA are thousands of times lower than the NRC Safety Goal and millions of times lower than the general U.S. cancer fatality risk.

WHAT ARE THE LIMITATIONS OF COMPARING SOARCA RESULTS TO THE NRC SAFETY GOAL AND OVERALL U.S. CANCER RISKS?

Comparisons of SOARCA's calculated long-term cancer fatality risks to the NRC Safety Goal and the average annual U.S. cancer fatality risk from all causes are provided to give context that may help the reader to understand the contribution to cancer risks from these nuclear power plant accident scenarios. However, such comparisons have limitations for which the readers should be aware. Relative to the safety goal comparison, the safety goal is intended to encompass all accident scenarios. SOARCA does not examine all scenarios typically considered in PRA, even though it includes the important scenarios. In fact, any analytical technique, including PRA, will have inherent limitations of scope and method. As a result, comparison of SOARCA's scenario-specific calculated long-term cancer fatality risks to the NRC Safety Goal is necessarily incomplete. However, it is intended to show that adding multiple scenarios' low risk results in the 1 in 1 billion to 1 in 100 billion range to approximate a summary risk from all scenarios, would yield a summary result that is also below the NRC Safety Goal of 2 in 1 million.

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

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 containments are usually cylindrical with a dome-shaped top and made of steel-reinforced concrete and a steel liner.

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.

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 weather conditions, 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 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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U.S. Nuclear Regulatory Commission

NUREG/BR-0359

January 2012

ATTACHMENT 44

K. Vierow, Y. Liao, J. Johnson, M. Kenton, and R. Gauntt, “Severe accident analysis of a PWR station blackout with the MELCOR, MAAP4, and SCDAP/RELAP5 Codes,” *Nuclear Engineering and Design 234*, at 129-145 (2004)

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Nuclear Engineering and Design 234 (2004) 129–145

Nuclear Engineering and Design

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Severe accident analysis of a PWR station blackout with the MELCOR, MAAP4 and SCDAP/RELAP5 codes

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Received 18 May 2004; received in revised form 19 August 2004; accepted 1 September 2004

Abstract

Based on knowledge obtained from experimental programs, considerable progress has been made in severe accident code development. The three leading severe accident codes used in the U.S., MELCOR, MAAP4 and SCDAP/RELAP5, are compared herein as part of an evaluation of the relative state of severe accident modeling in each of the codes.

The MELCOR code is evolving from a probabilistic risk assessment tool to a best-estimate severe accident system analysis code. Some advantages that MELCOR has are the capability to evaluate containment behavior and the source term to the environment, and the great modeling flexibility that the control volume approach and control functions afford. The MAAP4 code was developed to perform fast-running full simulations of severe accidents. Due in part to the simplified form of the conservation equations and the coarser discretization of the reactor systems, MAAP4 has calculation times far shorter than those of the other codes while producing credible results. The SCDAP/RELAP5 code contains more mechanistic physics models than the other codes for both severe accident and thermal-hydraulic phenomena and has undergone extensive validation against plant and experimental data.

The codes' overall attributes, relevant physics models and calculation results are compared herein. A hypothetical TMLB' scenario (station blackout with no recovery of auxiliary feedwater) at a 4-loop pressurized water reactor (PWR) was calculated by all three codes. Detailed plots showed that, despite considerable differences in the codes themselves, the calculation results of the codes are very similar in terms of thermal-hydraulic and core degradation response. There are minor discrepancies in various timings of phenomena, which are within the uncertainties of the code numerical computation and the physics models. The thermal challenge to the primary loop radionuclide barrier, such as the steam generator tube heat structures, is also compared among the three codes.

Abbreviations: LOCA, loss of coolant accident; PORV, power operated relief valve; PRA, probable risk assessment; PWR, pressurized water reactor; SG, steam generator; TMLB', station blackout with no recovery of SG auxiliary feedwater

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0029-5493/$ – see front matter © 2004 Elsevier B.V. All rights reserved.
doi:10.1016/j.nucengdes.2004.09.001

1. Introduction

The Laboratory for Nuclear Heat Transfer Systems at Purdue University is evaluating the current state-of-the-art in severe accident modeling by system codes in collaboration with severe accident code developers. MELCOR, MAAP4 and SCDAP/RELAP5 are the three leading severe accident simulation codes in the U.S. They have been developed from different approaches and for different purposes. A comparison of their respective physics models and calculations under conditions as consistent as possible are reported herein. The objective is to compare the three codes in a consistent and unbiased manner. The results of this comparison will be part of an assessment of where the codes stand with respect to each other, and to better understand the needs for severe accident modeling.

The codes have undergone significant upgrades over the years and are becoming more best-estimate in nature. MELCOR was originally intended to be a probabilistic risk assessment (PRA) tool. The initial objective for the MAAP4 code was to predict severe accidents, using simple models based on first principles. SCDAP/RELAP5 began as a best estimate code with physics-based models. Later versions of MELCOR contain significant modifications, including the addition of a large number of physics models. These changes render MELCOR capable of handling the more detailed spatial nodalizations typically used by SCDAP/RELAP5. MAAP4 has been shown to produce credible results for several severe accident scenarios despite relatively coarse spatial nodalizations and run times two or three orders of magnitude shorter than those of MELCOR and SCDAP/RELAP5. However, many aspects of MAAP4 are not publicly available, such as details of the models, and there is interest in why such a seemingly simpler code can perform so well.

While still quite different, the codes' capabilities and applications have also been converging over the years. MELCOR and SCDAP/RELAP5 are used by regulatory agencies and research institutions to evaluate several hypothetical severe accident events such as a station blackout or the potential for steam generator tube rupture (U.S. NRC, 1998). MAAP4 is the most widely used severe accident code by nuclear utilities and vendors because of its short run time and reduced requirements for code expertise. It is being used by the Electric Power Research Institute and many utilities for the Significance Determination Process and other analyses.

In the current work, the codes are compared with an emphasis on the severe accident modeling capabilities. The event chosen was a station blackout with no recovery of auxiliary feedwater to the steam generators (TMLB') for a 4-loop Westinghouse reactor based on Zion. This scenario provides an appropriate basis for a code comparison since the event progression tests a significant portion of the severe accident models in each code. Conservative analysis conditions necessary for investigation of the integrity of the steam generator tubes and other components were also used in the current analysis to ensure progression of the severe accident. Specifically, the reactor coolant system was not allowed to depressurize following any predicted pressure boundary failure, pump seal leakage was not considered and no operator or recovery actions were simulated.

The input decks for the three codes were made as similar as possible and consistent conditions were placed on all of the analyses. To arrive at a set of conditions that all of the codes could cover, various assumptions had to be made that rendered the calculation unrepresentative of plant behavior after some point in the calculation. Since the analysis scope of SCDAP/RELAP5 is limited to failure of the primary side pressure boundary, this comparison is focused on in-vessel severe accident phenomena, with a special interest in the steam generator tube response to thermal transients during this period.

It is emphasized that the objective herein is a comparison of the three codes, as opposed to a validation. Since extensive documentation of SCDAP/RELAP5 validation already exists, SCDAP/RELAP5 input and results are assumed to be the reference in many aspects of the calculation herein; a few qualifications regarding the current state of SCDAP/RELAP5's modeling are noted later. Plans for code validation against plant and experimental data will be discussed as future work.

2. Overview of the codes

MELCOR, MAAP4 and SCDAP/RELAP5 are the leading codes in the U.S. for simulating severe accident events. All three codes are capable of modeling reactor coolant system response, core material chemical reactions, core heat-up, degradation and relocation, heat structure response, and other severe accident phenomena. Modeling of fission product release and transport and containment phenomena are integrated into the MELCOR and MAAP4 codes.

SCDAP/RELAP5 is characterized by its detailed, mechanistic models of severe accident phenomena, however, the calculations can be rather time-consuming. SCDAP/RELAP5 typically uses on the order of hundreds of hydrodynamic components to model the primary system. MAAP4 calculations require minimal computation time with simplified geometry models. MELCOR falls in between these two codes, being much closer to SCDAP/RELAP5 in terms of nodal complexity. MELCOR runs at a moderately fast speed and has a large number of mechanistic models. In early applications, MELCOR's spatial discretization of a nuclear power plant consisted of roughly 10–30 control volumes and a large number of parametric calculations could be run in a short time. With the more complicated calculations that are now being demanded of it, MELCOR's run times have been increasing. The problem complexity and runtimes are roughly equivalent to those of typical SCDAP/RELAP5 calculations. The code developers are hence incorporating numerical methods that are more robust and efficient in handling systems that are larger than originally intended.

2.1. Highlights of MELCOR

MELCOR version 1.8.5 is used herein and is the latest released version (Sandia National Laboratories, 2000a, 2000b).

Although the MELCOR code was originally developed as a PRA code by Sandia National Laboratories, contemporary applications are largely detailed best estimate calculations. An advantage that MELCOR has over other the codes is the great modeling flexibility that the control volume approach and control functions afford. Virtually any reactor geometry may be described in the input. MELCOR is also able to perform integral analysis of reactor systems since it has the capabilities to model the containment response and fission product behavior out to the source term.

A powerful feature of MELCOR, "sensitivity coefficients" provide the user with the capability to easily change a large number of modeling parameters via input and perform sensitivity studies. This feature, however, places additional demands on the user because user expertise is required to determine appropriate values for the coefficients and little guidance is available in determining reasonable values.

2.2. Highlights of MAAP4

The MAAP4 code models severe accidents in LWRs following large and small break Loss of Coolant Accidents (LOCAs) and transients. Developed for the Electric Power Research Institute (EPRI) by Fauske et al. (1994a, 1994b, 1994c), the latest available version, MAAP4.05, is used herein.

MAAP4 employs "generalized models" for BWRs and PWRs, in which the type and number of components and the geometry are predetermined. The user inputs various parameters for each component such as volumes or masses. Similar to MELCOR, the MAAP4 code can perform integral analysis of reactor systems, however, the standard version is limited to conventional light water reactors. Code versions have been created to model other NSSSs including CANDU and VVER designs and advanced reactors.

2.3. Highlights of SCDAP/RELAP5

The latest version of SCDAP/RELAP5 is a combination of the RELAP5 MOD 3.3 code for thermal hydraulics, SCDAP for the severe accident-related phenomena and the COUPLE code for a finite element treatment of the vessel lower head (Siefken et al., 2001a, 2001b). The code was developed by Idaho National Engineering Laboratory (INEL) for LOCAs and transients within the primary system, reactor core phenomena and fission product release from the fuel.

Mechanistic models for thermal hydraulics and many severe accident phenomena have been extensively verified and documented against experimental and plant data to a greater extent than for the other two codes. The number of modeling parameters that must be determined by the user is less than in MELCOR, reducing the required expertise level of the user.

A drawback is that calculations cannot be performed beyond pressure boundary failure of the primary side. The recent version of SCADAP/RELAP5 has omitted the fission product behavior models from its previous version and left this aspect of the calculation and the containment performance to be analyzed by the VICTORIA (Bixler and Schaperow, 1998) and CONTAIN (Bergeron et al., 1985) codes.

2.4. *Thermal-hydraulic models*

The thermal-hydraulic modeling is mentioned briefly here because the results of these models strongly affect the severe accident progression. SCDAP/RELAP5 employs the detailed mechanistic models of the RELAP5 MOD 3.3 code for single and two-phase flow. The one-dimensional, semi-implicit forms of the conservation equations for the two-fluid model are solved. The robust RELAP5 modeling is clearly superior to the thermal-hydraulic models of the other codes. MAAP4 has simplified but fast-running models for thermal hydraulics description using a fixed nodalization of the primary circuit. MAAP4 solves a set of lumped parameter, first-order differential equations for conservation of mass and energy. Differential equations for momentum conservation are not employed because MAAP4 considers momentum balances to be quasi-steady, which reduces the momentum equations to algebraic equations. MAAP4 has a highly simplified treatment of the natural circulation of single-phase water and two-phase steam/water mixtures in the reactor coolant system, with the level of modeling detail increasing substantially during the time frame leading up to the uncovering of the core. MELCOR lies closer to SCDAP/RELAP5 in terms of complexity and accuracy of the two-phase flow modeling. Semi-implicit finite difference equations are solved for conservation of mass, momentum and energy and mechanistic models are employed depending on the phenomena.

2.5. *Core models*

Generally, the early phase of a hypothetical severe accident is dominated by thermal-hydraulic phenomena. Sophisticated thermal-hydraulic modeling is important for accurate prediction of the early reactor response and timing of events henceforth. On the other hand, the later phases of severe accidents are dominated by additional complicated phenomena in addition to the thermal-hydraulics. Severe accident models of the codes are more diversified, less validated, and hence have more uncertainties.

The manner in which core materials are handled has a large effect on code calculation results. The MELCOR and SCDAP/RELAP5 codes have a thermal-hydraulic nodalization for the primary system and model the core as structures within this nodalization. For the thermal-hydraulic nodalization, the reactor vessel is discretized into radial rings and axial levels. Recent applications of a 3-D version of SCDAP/RELAP5 have been presented in which the rings are further divided into segments (Coryell et al., 2002). Core structures in both codes are placed within the thermal-hydraulic control volumes. The axial division of SCDAP/RELAP5 thermal-hydraulic control volumes for the core region is often somewhat finer than in MELCOR. However, the core cells are of similar sizes because the MELCOR core structures typically have a finer axial nodalization than the MELCOR thermal-hydraulic control volumes.

The MAAP4 approach is to include the coolant in a core "node" along with the core materials. Core node sizes are comparable to those in the other codes.

Another significant difference is the detail in which the core transitions from an intact state to a degraded state. The models of core material relocation in the three codes have substantial differences. The most detailed models for material interactions and relocation have been incorporated into SCDAP/RELAP5. MELCOR does not treat liquefied core materials in detail. If not associated with a molten pool, any liquefied material is assumed to relocate to a lower cell and resolidify on a solid component during the current timestep. MELCOR estimates where the frozen corium will reside based on heat transfer and other considerations and places the melt at this location within the current timestep. MAAP4 does explicitly model the motion of liquefied core materials but lumps all degraded components in a given node together. The lumped materials are assigned the same temperature, although temperature gradients are algebraically calculated across peripheral core nodes to compute crust thicknesses and heat transfer rates.

As there is a lack of advanced understanding about some core material interaction and relocation phenomena, it is tempting to say that simplified or parametric core degradation models can do as well as detailed, mechanistic models, considering that large uncertainties are involved in the models. MELCOR and MAAP4 originally resorted to simplified or parametric models, which are adequate for PRA calculations. The earlier models have been superceded by more mechanistic treatments that render the codes capable of best-estimate calculations.

Although substantial differences exist in three codes' core models, some comparison results are surprisingly consistent. For instance, the total hydrogen production, the total molten core material mass, and the total core debris mass slumping into lower head are quite similar for three codes' predictions during the in-vessel severe accident phase. This comparison will be further discussed in Section 5.

3. Reference calculation

A hypothetical TMLB' scenario at a Zion 4-loop PWR was chosen as the reference calculation. A TMLB' event is a station blackout where there is a loss of A/C power and no recovery of the steam generator auxiliary feedwater supply. This complex event has many variables that change the progression of the accident and is a good test of a large number of severe accident models. It is assumed herein that there is no operator or outside intervention and pump seal failures do not occur.

Originally built in the early 1970s, the two-unit Zion nuclear power plant began operation in 1973 and 1974 and both units were shut down permanently in 1998. The reactors are Westinghouse 4-loop PWRs. Units I and II have net capacities of 1040 MWe each for a total plant capacity of 2080 MWe. Zion was chosen for this comparison as it is a representative plant for 4-loop PWRs.

The event begins with the loss of A/C power and steam generator feedwater supply. The turbine immediately trips and the reactor SCRAMs. The water in the steam generator secondary side boils away, resulting in a loss of heat sink. As the primary-side pressure increases, the pressurizer Power Operated Relief Valves (PORVs) open. (The PORVs are assumed herein to operate on battery power.) This LOCA results in a decrease of primary-side inventory.

Natural circulation of hot gasses is predicted to begin when the primary-side water level has decreased below the top of the core. Due to the very hot steam (and later, hydrogen) coming out of the vessel, there is the potential for excessive heatup of structural components such as hot leg nozzles, the surge lines, and steam generator tubes, along with the potential failure of these pressure boundaries. If the pressure boundaries are compromised, radioactive material may be released to the containment or environment. Core heatup and uncovery result in fuel failure. Molten fuel may then relocate to the lower plenum and damage the lower head and/or penetrate the vessel. Upon failure of the lower head, molten material may be released to the cavity. Ex-vessel phenomena may include molten fuel-coolant interactions, corium-concrete interactions and resulting containment pressurization.

For the current work, in addition to the code comparison, a second objective is to investigate the integrity of the steam generator tubes. To be conservative, the relief valves on the pressurizer loop steam generator are assumed to fail upon first challenge and remain stuck open thereafter. This results in boiloff of the secondary side and a large pressure differential across the steam generator tube walls. Depressurization following any heat structure failure is artificially suppressed to extend the code comparison and to investigate the order of heat structure failures, recognizing that the reactor response is not representative beyond the time of the first failure.

One of the most challenging aspects of the event for the codes to reproduce is natural circulation cooling of the system. In addition to the vessel-to-steam generator path, natural circulation within the vessel can develop, driven by heat loss from the vessel walls. If the water seal in two cold legs, or in one cold leg and the base of the core barrel is cleared, steam can also make a full loop from the vessel to the steam generators and return through the cold leg. Essentially, the steam would take the same path that the water coolant would take during normal operation. This would reduce the temperature difference between the hot legs and steam generator tubes, increasing the thermal challenge to the latter.

4. Method of comparison

4.1. Codes versions

The code versions used in this work are MELCOR version 1.8.5, MAAP4.0.5 and SCDAP/RELAP5 MOD 3.3. For a more level comparison basis, MELCOR and SCDAP/RELAP5 were compiled on a PC with the same compiler and compiler options at Purdue University. Since the MAAP4 code is not available to Purdue University, the MAAP4 calculations were conducted at Creare, Inc.

4.2. Input preparation

Input preparation for the three codes was made to minimize input discrepancies. Due to different code application practices, code model requirements and purposes, each code has some unique input.

All three of the codes' inputs are based on plant data. The base MELCOR input deck for a Zion station blackout was provided by Sandia National Laboratories. The loop with the pressurizer is modeled as a single loop while the other three loops are grouped together and modeled as one. The thermal-hydraulic nodalization of the vessel core region is a 5-ring, 4-level control volume geometry (Fig. 1). The core nodalization is represented as a 5-ring, 12-level model with three core control volumes per thermal-hydraulic level and 10 heated levels.

The MAAP4 model represents the reactor coolant system with 15 nodes. Two loops are modeled, a combined loop containing three steam generators and a pressurizer loop containing the fourth. The core itself is modeled by four rings, with 13 axial levels. Gas temperatures and compositions in each of these core nodes are computed quasi-statically, and the gas emerging from the core is added to a single thermal-hydraulic node that represents the core volume as a whole.

All four loops are individually modeled in the SCDAP/RELAP5 deck and several piping components have a finer nodalization than in the MELCOR and MAAP4 models. The three intact loops are observed to behave identically, justifying the MELCOR and MAAP4 approach of grouping them together. The SCDAP/RELAP5 core has a 5-ring, 10-level nodalization.

SCDAP/RELAP5 provides for a very detailed nodalization of the lower plenum and lower head by the COUPLE code, while MELCOR's modeling uses a small number of lower plenum control volumes and a four-temperature node conduction model through the vessel wall. Although the MAAP4 lower plenum model is coarsely nodalized, it represents relatively detailed phenomena in order to predict the timing and mode of vessel failure. The MAAP4 model allows the various constituents of the debris to exist as particulates, crust layers on structures, or as components of a bed consisting of a metal layer and an oxidic debris pool.

4.3. Modeling of natural circulation

To compare the late phase natural circulation and the thermal transient on heat structures that act as a primary loop radionuclide barrier, some input and model options that were not standard had to be employed. Two issues in input preparation required special attention.

The first issue is to capture the natural circulation flow paths. As shown in Fig. 1, for MELCOR, the hot leg is modeled in two sections, an upper half and a lower half. The halves are connected by flow paths, which allow mixing between them. Such a division of the hot leg is necessary to model the flow of steam to and from the steam generators because countercurrent flow of a single fluid cannot be calculated in a single control volume. The SCDAP/RELAP5 input deck includes a similar nodalization scheme and MAAP4 uses a similar approach.

The second issue is the recirculation ratios in the steam generator primary side and reactor vessel upper plenum. As shown in Fig. 2, during the natural circulation phase, multiple recirculation flow patterns may develop. The experimental data as well as computational fluid dynamics (CFD) analysis results show that the steam generator recirculation ratio controls the cooling capability of the steam generators and greatly influences the temperature difference between the hot legs and the steam generator tubes (Boyd et al., 2003). Prediction of the actual steam generator recirculation ratio is performed only by MAAP4. A detailed discussion of the natural circulation and the definition of the recirculation ratios are provided in Section 5.

Although stated previously, it is emphasized here that the objective of this study is a comparison of the three codes, as opposed to a validation. In order to provide an unambiguous comparison, the following approach was taken. First, input parameters were adjusted to yield similar initial values for the hot leg natural circulation flow rate. Second, the MAAP4 model for steam generator/hot leg recirculation flow ratio was over-ridden so that the model would yield approximately the same initial value for this parameter as is assumed in the other two codes. Third, the pressure drop caused by cross-flow in the core was neglected in MAAP4; this yielded a time-dependent core/upper plenum natural circulation flow rate that was very similar to the other two codes. Fourth, the MAAP4 model for thermal radiation between gasses and structures in

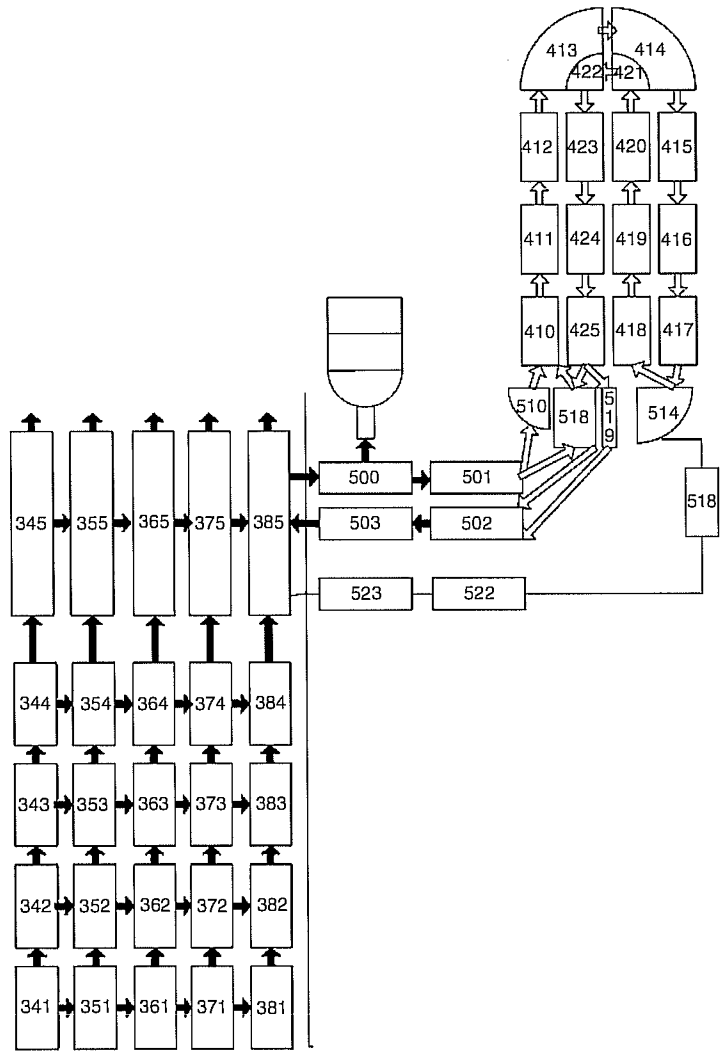

Fig. 1. MELCOR nodalization for natural circulation.

the hot leg and surge line was disabled since this process is not represented in the other codes. Finally, the hot leg creep rupture calculation was configured in all three codes to use properties corresponding to the hot leg piping, rather than the hot leg nozzle safe end. Based on previous work (Bayless et al., 1995; EPRI, 2002), the last two assumptions will tend to substantially delay the timing of hot leg creep rupture relative to surge line and tube failure. Thus, the predicted results should only be used for comparing model predictions and should not be considered to reflect what would actually fail first.

5. Comparison of calculation results

5.1. System thermal-hydraulic response

All three codes calculate a similar thermal-hydraulic response for the primary and secondary loops. Figs. 3 and 4, respectively, show the pressure response for the primary loop and for the secondary pressurizer loop with the safety relief valves stuck-open.

The primary-side pressure transient exhibits a cyclic pattern (Fig. 3) caused by the cycling of the pressurizer PORVs. The pressurizer pressure initial dip within the

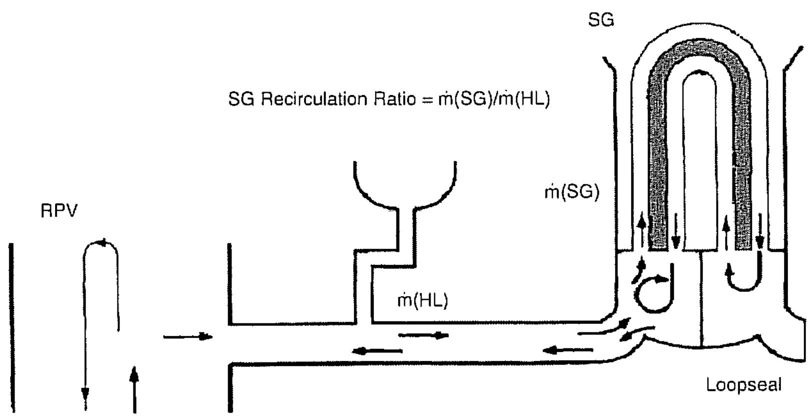

Fig. 2. Natural circulation flow patterns.

first 2000 s is due to a high heat removal rate from the steam generator on the pressurizer loop with a stuck-open relief valves. Enough energy is being removed that the primary side pressure is reduced. MELCOR and SCDAP/RELAP5 predict this pressure dip similarly. The reason for MAAP4 not predicting this dip is unclear, since this is usually seen in MAAP4 simulations of transients in other plants.

The boil-off pattern of the secondary side (Fig. 4) confirms the similarity of the predictions of the heat transfer models in the three codes. Also shown by Fig. 4, the good agreement between MELCOR and SCDAP/RELAP5 calculations is very apparent. MAAP4 shows fairly similar trends during the initial pressurization but has a slightly different profile between 1000 and 2000 s.

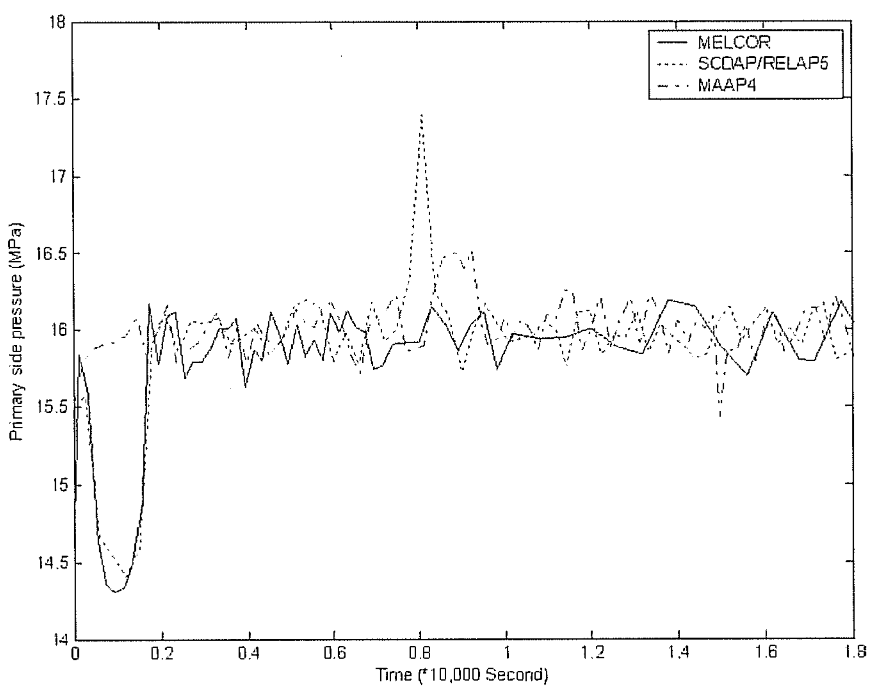

Fig. 3. Primary side pressure.

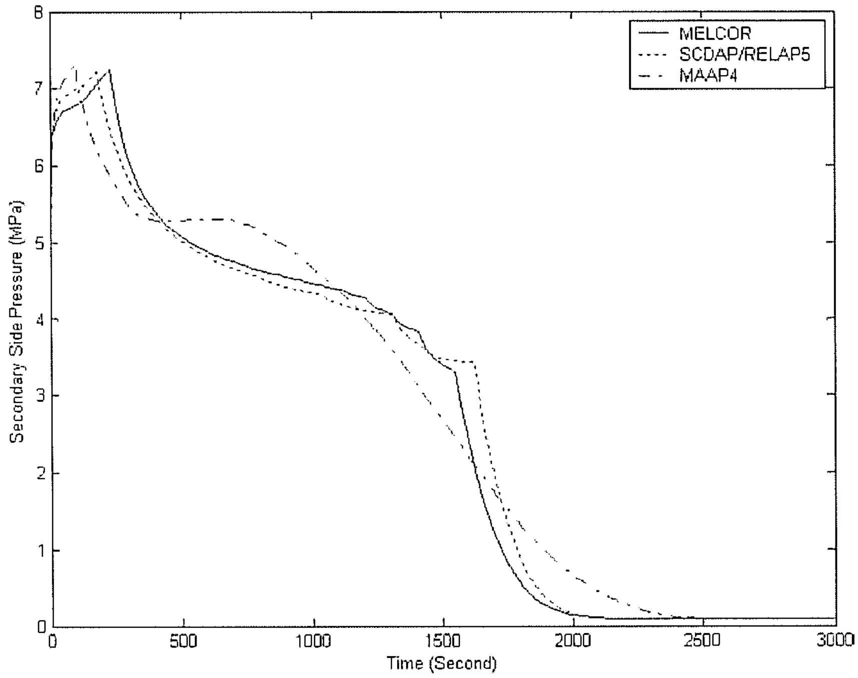

Fig. 4. Secondary side pressure.

As shown in Fig. 5, the three codes calculate a similar pressurizer collapsed water level. The water level decrease before 2000 s in the MELCOR and SCDAP/RELAP5 results reflects the primary-side pressure dip. Following this period, the gas phase in the pressurizer is quickly vented through the PORVs and the liquid phase is pushed upward to occupy the pressurizer volume completely. After core uncovery (Table 1), a large amount of steam is generated from boiling in the core. If the steam volumetric generation rate is greater than the liquid volumetric venting rate through the PORVs, the primary system may experience a steep pressure rise, as seen in the SCDAP/RELAP5 results of Fig. 3 at about 8000 s. Such a phenomenon is predicted to some extent by MAAP4 but not by MELCOR, since during this potential period of primary pressure increase, MELCOR calculates a greater venting flow rate through the PORVs.

5.2. *Fuel and core damage*

The most challenging task in comparing the three codes deals with fuel and core damage. The damage is dependent not only on the core coolant flow patterns determined by thermal-hydraulic models, but also on the metal oxidation model, core material interaction and

Table 1
Timing of key events

Event	MELCOR (s)	SCDAP/RELAP5 (s)	MAAP4 (s)
Start of core uncovery	7,680	7,160	9,615
Core completely voided	11,620	9,950	14,500[a]
5% cladding oxidized	13,780	14,806	13,800
Slumping to lower head	16,189	16,130	21,994

[a] MAAP4 calculated a very slow rate of water level decrease at the bottom of the core, leading to a substantial delay in the voiding of the bottom node. This is, in part, due to continued slow draining of the pressurizer, as shown in Fig. 5.

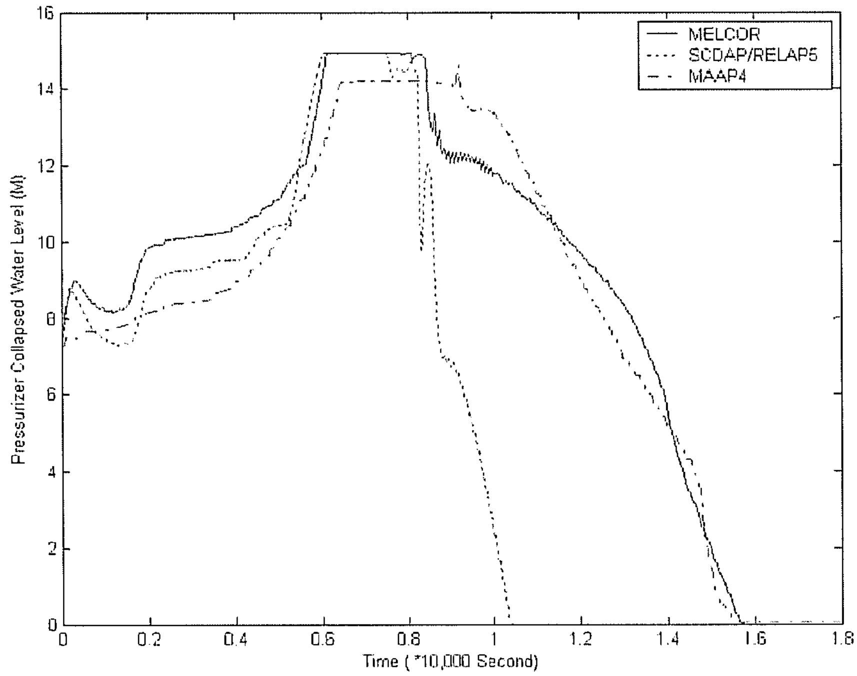

Fig. 5. Pressurizer collapsed water level.

transport model, and other models. Table 1 compares some events occurring during the in-vessel fuel and core damage period. MELCOR and SCDAP/RELAP5 predict similar timings for core uncovery and core degradation events. MAAP4 predicts a later timing for the start of core uncovery, which delays the timing of subsequent events. This difference is partially explained by the fact that core uncovery in the former codes is based on the collapsed level, whereas in MAAP4 the two-phase level is used. Core slumping is also notably later in MAAP4.

MELCOR has good capabilities for tracing masses separately for several types of core materials and for different phases of the same kind of material such as intact fuel, particular debris or conglomerate debris. By following the transport of individual core materials among core cells and the change from one physical phase to another within MELCOR, the core degradation progress can be constructed. The timing for core slumping to the lower head is determined by the timing of core support structure failure.

SCDAP/RELAP5 and MAAP4 appear to model a molten pool as that occurred in the TMI-2 accident. The molten pool is enlarged by acquiring molten materials in neighboring core cells. Failure of the crust surrounding the molten pool triggers the molten pool slumping to the lower head.

Hydrogen production accompanying fuel and core damage is a good comparison index, since it represents an integrated effect of different models on the core degradation progress. Fig. 6 shows hydrogen production as calculated by the three codes.

The onset of hydrogen production, at about 11,000 s, is almost identical for the three codes. The hydrogen production rate depends on several conditions including the metal oxidation model, the availability of coolant for chemical reaction and the fraction of metal surface exposed to coolant. Both MELCOR and SCDAP/RELAP5 apply parabolic kinetics for the high temperature oxidation model. MELCOR starts calculating the parabolic oxidation at a lower temperature than SCDAP/RELAP5 with the Urbanic–Heidrich correlation. Thus, before 14,750 s, MELCOR calculates more hydrogen production than SCDAP/RELAP5. However, after 14,750 s, the core temperatures are high enough that SCDAP/RELAP5 starts calculating

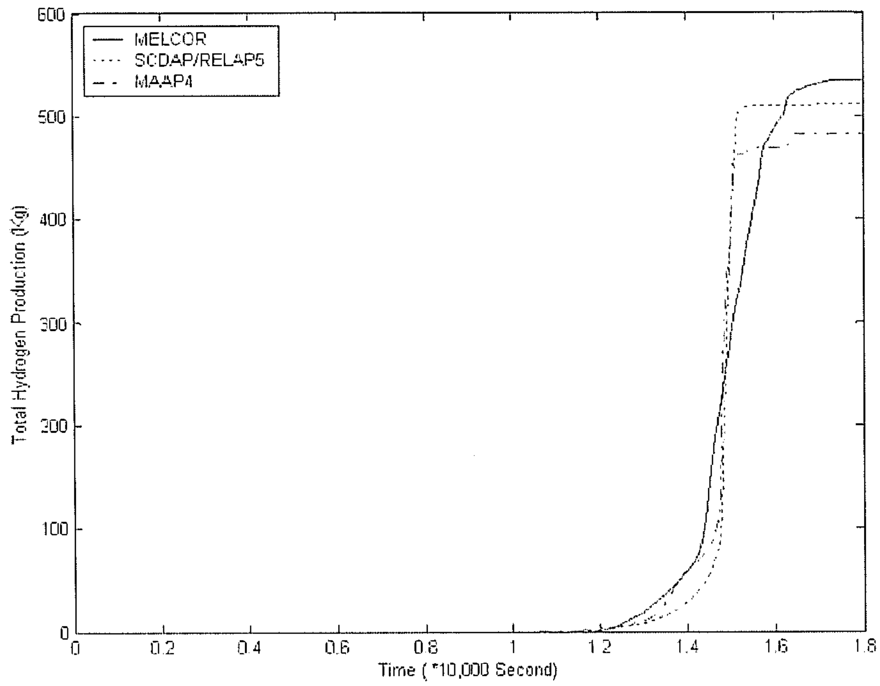

Fig. 6. Hydrogen generation.

the parabolic oxidation. SCDAP/RELAP5 uses the Cathcart and Pawel correlation up to a temperature of 1853 K. This correlation predicts a greater hydrogen generation rate than would be predicted at the same temperature by the Urbanic–Heidrich correlation employed by MELCOR. Therefore, SCDAP/RELAP5 predicts greater hydrogen and chemical reaction heat rates.

As the molten core slumps into the lower head, additional hydrogen is produced with MELCOR, but little is produced with SCDAP/RELAP5 or MAAP4. This difference may be caused by a different core debris geometry when slumping to and residing at lower head and different code treatments (Leonard et al., 1996).

5.3. Creep rupture of primary loop pressure boundary heat structures

As the liquid phase is vented out of the PORVs from the pressurizer and the hot leg after core uncovery, a counter-current natural circulation loop is established, as shown in Fig. 2. The nodalization for modeling this flow pattern in MELCOR is provided in Fig. 1. Hotter steam flows from the reactor vessel through the upper part of the hot leg to the steam generator inlet plenum. After mixing with steam in the steam generator inlet plenum, some of the steam enters the steam generator 'hot' tubes. Cooler steam returning from the steam generator outlet plenum flows to the reactor vessel via the lower part of the hot leg.

During this period, the steam produced by core decay heat and metal–water reaction is hot enough to challenge the integrity of the primary loop boundary heat structures, such as the steam generator tubes, pressurizer surge line and hot legs. In the station blackout scenario, this causes a risk of failure at the primary loop boundary heat structures before the reactor vessel lower head breach. The worst scenario would be a rupture of the steam generator tubes, resulting in a direct containment bypass of radioactive materials to the outside environment. Therefore, depressurization is prevented following prediction of a failure and the order of failures is compared.

Due to the limited capability of all three codes to model the three-dimensional problem and complicated flow patterns, a mixing parameter called the "recircu-

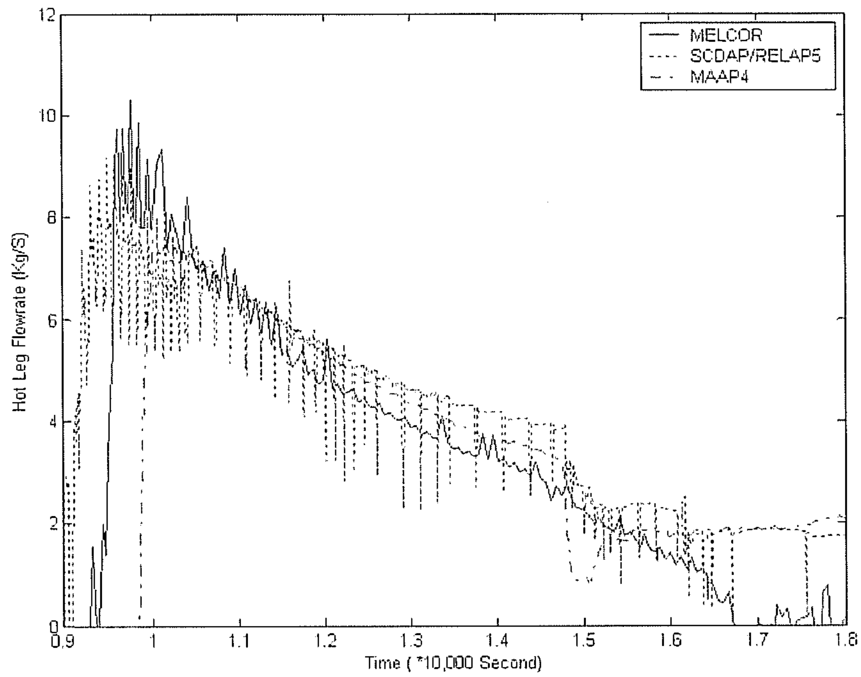

Fig. 7. Hot leg flow rate in pressurizer loop.

lation ratio" is used to regulate the natural circulation flow rates in SCDAP/RELAP5 and MELCOR. This recirculation ratio is defined as the flow rate from the steam generator inlet plenum to the steam generator hot tubes divided by the flow rate entering the steam generator inlet plenum from the upper part of the hot leg. Experimental data from the Westinghouse one-seventh scale test facility shows that this recirculation ratio is about 2 (Boyd et al., 2003). An analysis study with CFD has also confirmed this result (Boyd et al., 2003). As stated earlier, MAAP4 contains a model for predicting the recirculation ratio, but this model was over-ridden in order to allow a direct comparison to the other codes' predictions. Figs. 7–9 compare the flow rate entering the hot leg, the steam generator tube flow rate and the recirculation ratio, respectively for the loop with the pressurizer.

The steam generator recirculation ratio can affect the thermal transient on the steam generator tubes, pressurizer surge line and hot leg heat structures. A larger recirculation ratio value indicates more significant flow mixing in the steam generator inlet or outlet plenum, resulting in lower temperature steam entering the steam generator hot tubes and inhibiting the steam generator tube rupture. This in turn increases the failure risk of other primary loop boundary heat structures such as the pressurizer surge line and the hot leg. The steam generator recirculation ratio is not the single factor affecting heat structure creep rupture. Other system-wide factors include the thermal-hydraulic models, cladding oxidation model and core degradation model. With similar natural circulation flow patterns and recirculation ratios as shown in Figs. 7–9, the three codes' predictions for heat structure failure due to creep rupture are compared in Table 2.

In all three calculations, peak steam generator stresses were increased by 50% to crudely model the effects of corrosion and tube defects. As shown in Table 2, none of the codes predict that the first failure will be a steam generator tube rupture. The surge line was the first failure followed by either the steam generator tubes or the hot leg piping.

The Larson–Miller creep rupture failure model is used by all three codes to calculate the thermal transient on the heat structures. Heat structure creep failure depends on the pressure and temperature history. As

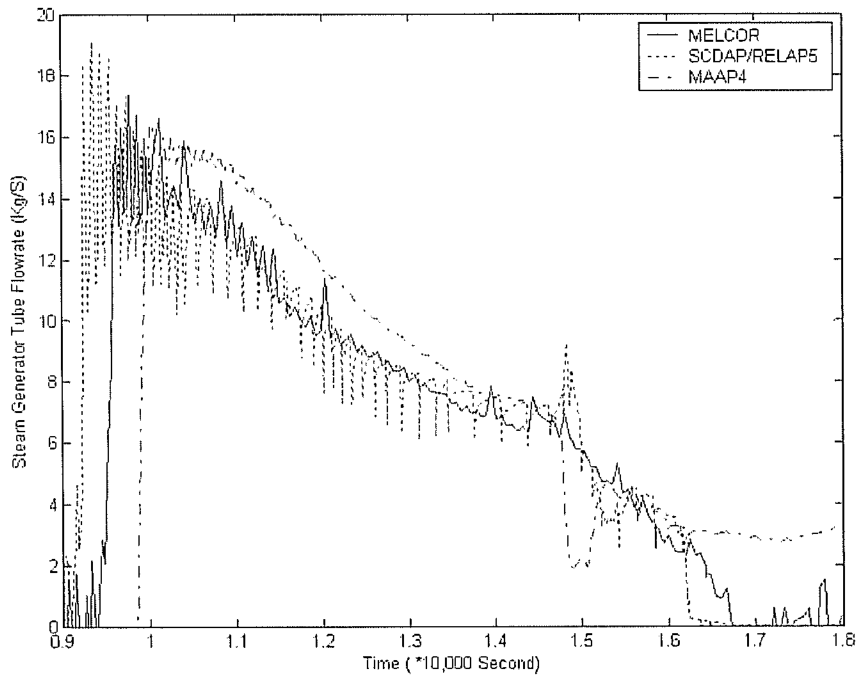

Fig. 8. Steam generator tube flow rate in pressurizer loop.

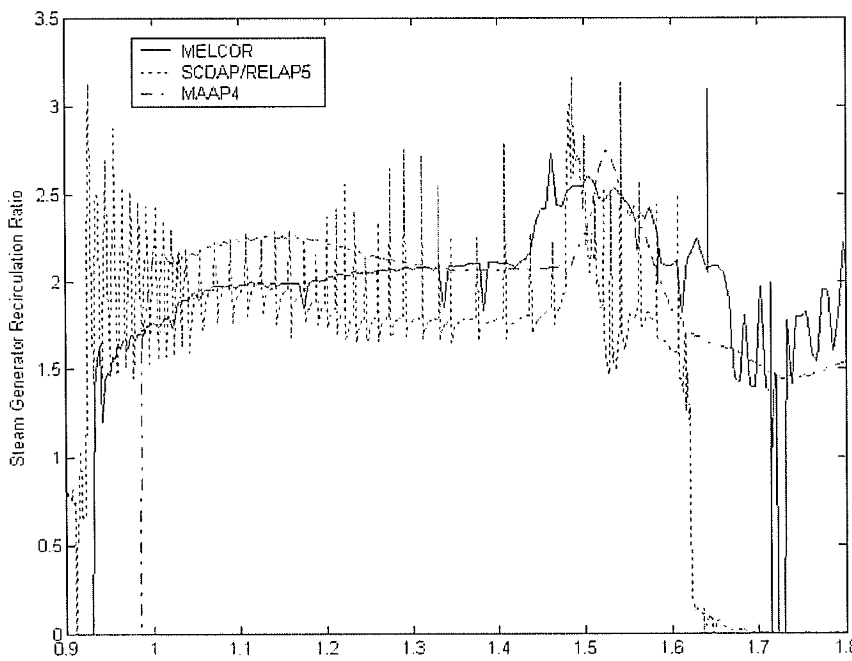

Fig. 9. Steam generator recirculation ratio in pressurizer loop.

Table 2
Timing of heat structure failures

Event	MELCOR (s)	SCDAP/RELAP5 (s)	MAAP4 (s)
Onset of natural circulation	9,300	9,000	9,720
Failure of surge line	16,287	14,955	14,860
Failure of hot leg piping on pressurizer loop	16,464	15,720	15,267
Failure of SG tubes on pressurizer loop	16,553	15,210	14,913

explained for the system thermal-hydraulic response, the pressure histories are similar in all three codes. Therefore, the variation in the creep failure timing is principally due to the heat structure temperature transients. The temperature transients of the surge line heat structure, hot leg heat structure and the steam generator tubes are shown in Figs. 10–12, respectively.

The heat structure temperatures closely overlap during the early phase of the accident. All three codes predict a similar temperature rise after onset of natural circulation, though with varying rising slopes. While temperatures are rising, all three codes predict the highest temperature in the surge line (Fig. 10) and the lowest temperature in the steam generator tubes (Fig. 12). This is consistent with the modeling of the natural circulation and the greater thickness of the hot leg. Hotter steam from the core flows directly to the surge line via the upper part of the hot leg. Some of this hotter steam is cooled by mixing in the steam generator inlet plenum before entering the steam generator tubes. Cooler steam returns from the steam generator to the lower part of the hot leg and mixes with hotter steam there.

During this period of thermal challenge to the heat structures, core decay heat power is identical for the three codes. As explained in the fuel and core damage section, due to a variation in metal oxidation models, SCDAP/RELAP5 calculates more hydrogen production and chemical reaction heat than MELCOR, resulting in a faster temperature rise in the heat structures.

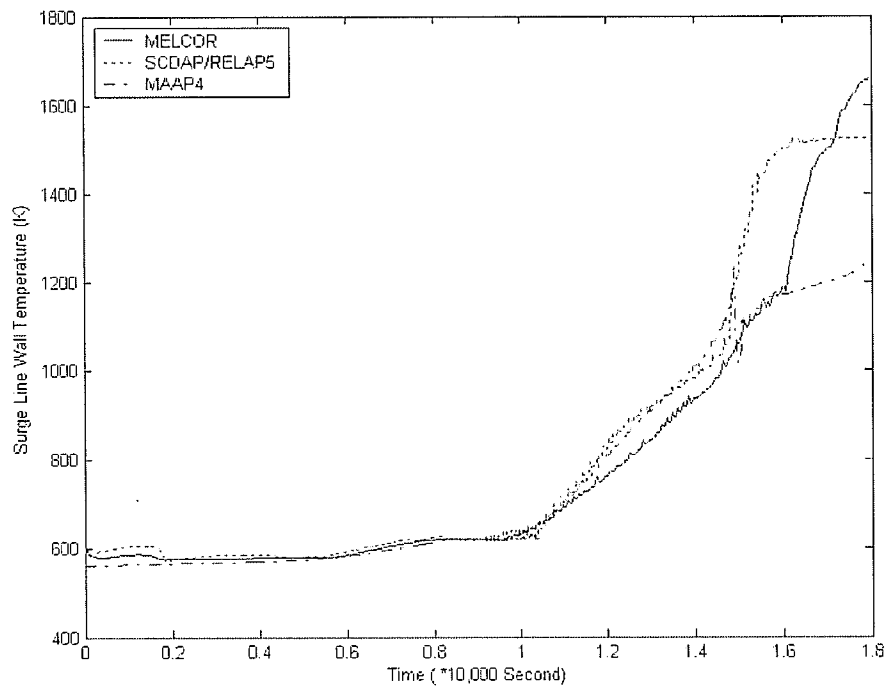

Fig. 10. Surge line wall temperature in pressurizer loop.

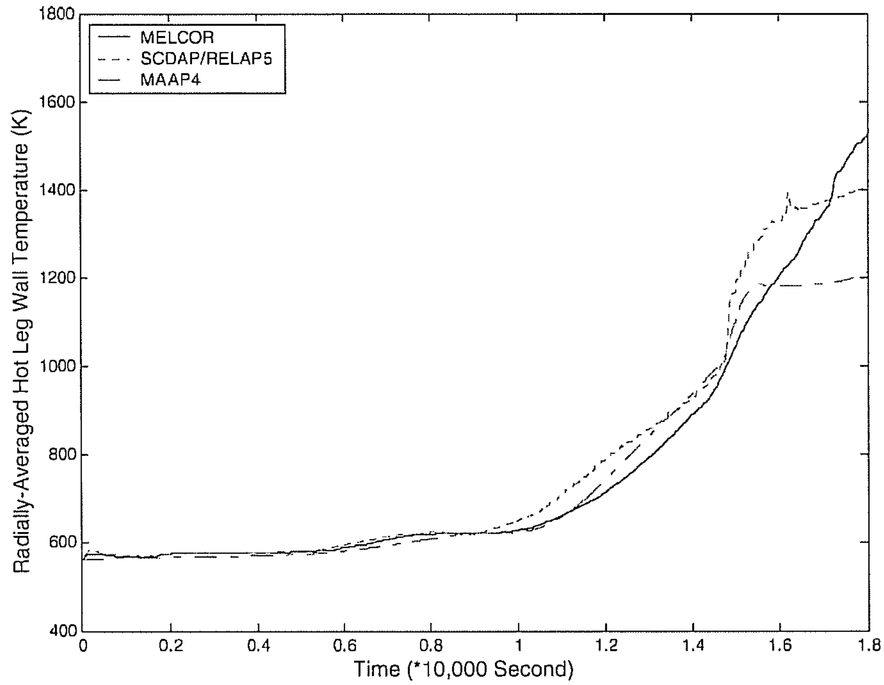

Fig. 11. Radially-averaged hot leg wall temperature.

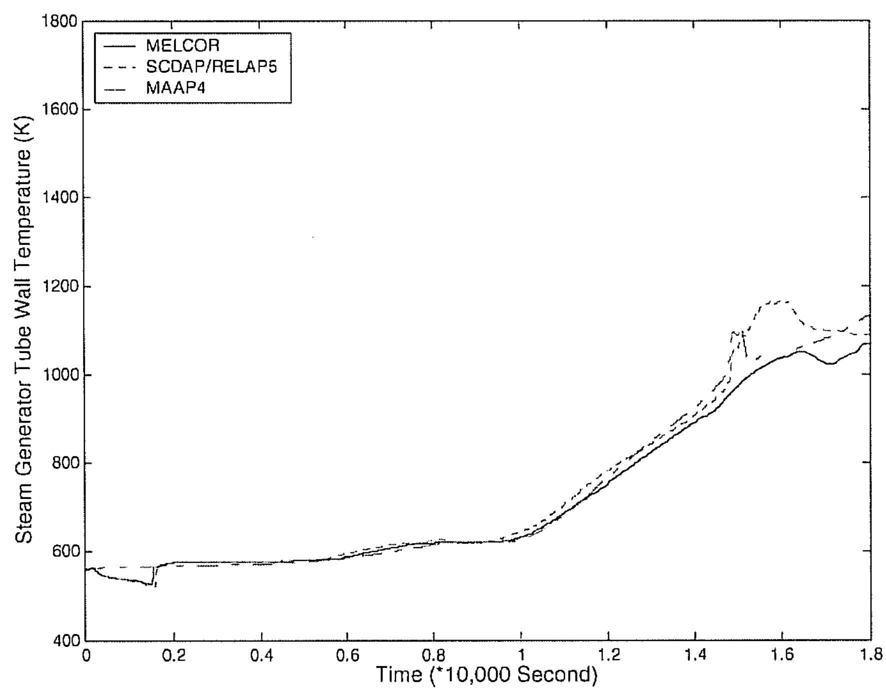

Fig. 12. Steam generator tube wall temperature.

6. Conclusions

Three severe accident analysis codes, MELCOR, SCDAP/RELAP5 and MAAP4 were compared for their thermal-hydraulic models and core degradation models. Calculations were performed for a hypothetical TMLB' accident at the Zion nuclear power plant. Efforts were taken to minimize differences in the code input, scenario assumptions, and model options to achieve an unbiased comparison for the codes performance.

Detailed plots show that the thermal-hydraulic phenomena and major in-vessel severe accident phenomena are in good agreement for the three codes. The integral effect of diversified core models in terms of total hydrogen production and total core debris mass slumping into reactor vessel lower head are consistent for the three codes. Though not discussed in the current work, it is expected that this consistency will reduce the codes' prediction differences for ex-vessel severe accident phenomena such as ex-vessel corium–water or corium–concrete reaction, hydrogen behavior in the containment and containment pressure response. There are several discrepancies that could be termed as minor and that are possibly due to uncertainties in the numerics and physics models.

Several key assumptions were made to account for known differences in heat transfer modeling and the representation of counter-current natural circulation of hot gasses. Given these assumptions, the three codes' predict similar temperatures in the various reactor coolant system components. Thus, future work could focus on resolving the modeling differences in these few key areas.

So far, plant data for actual severe accidents exists only for the TMI-2 accident. Severe accident code simulations for the TMI-2 scenario have been carried out with SCDAP/RELAP5 (Hohorst et al., 1994), MELCOR (Gauntt et al., 2002) and MAAP4 (Paik et al., 1995). Comparing the different severe accident codes' predictions against plant data is an essential test of the codes' accuracy and will provide additional information on the relative merits of the various severe accident models. In future work, code validation against plant data and experimental data will be performed to evaluate the correctness of the codes.

Acknowledgments

The authors would like to thank the U.S. Nuclear Regulatory Commission and the Electric Power Research Institute for their sponsorship and the use of the codes.

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ATTACHMENT 45

International Atomic Energy Agency, *IAEA International Fact Finding Expert Mission of the Fukushima Dai-ichi NPP Accident Following the Great East Japan Earthquake and Tsunami* at 33-35 (June 2011)

IAEA
Original English

MISSION REPORT

THE GREAT EAST JAPAN EARTHQUAKE
EXPERT MISSION

IAEA INTERNATIONAL FACT FINDING EXPERT MISSION OF THE FUKUSHIMA DAI-ICHI NPP ACCIDENT FOLLOWING THE GREAT EAST JAPAN EARTHQUAKE AND TSUNAMI

Tokyo, Fukushima Dai-ichi NPP, Fukushima Dai-ni NPP and Tokai Dai-ni NPP, Japan

24 May – 2 June 2011

IAEA MISSION REPORT

DIVISION OF NUCLEAR INSTALLATION SAFETY

DEPARTMENT OF NUCLEAR SAFETY AND SECURITY

Alignment of the valves to vent the Unit 2 containment was carried out on 13 March by opening an air operated valve using an air cylinder and another valve with AC power supplied by an engine generator. After the Unit 3 explosion, discussed below, the valve was rendered inoperable. The operators then attempted to open another air operated valve to establish the vent path. An engine driven air compressor and AC power supplied by an engine generator were used and the valve appeared to open slightly. However, the successful venting of the Unit 2 containment could not be verified.

After the HPCI failed in Unit 3 on 13 March, approximately seven hours elapsed until an alternative injection source could be established. The RPV pressure was reduced through steam discharge through one of the SRVs into the suppression pool. The accumulator of the SRV contained adequate nitrogen pressure so the SRV could be opened with car batteries. Once pressure was reduced, injection of water was established using a fire engine pump injecting through the fire protection and MUWC lines connected to the LPCI (one mode of the RHR system) lines. Boron was added intermittently. The suction of the pump was changed to a pit filled with sea water at one point temporarily interrupting injection for a short time, on the order of minutes. A further interruption occurred for two hours. Once restarted, a total of 4495 tonnes of sea water was injected from 13 March until 25 March, at which time a fresh water source was established similar to that of Unit 1.

Alignment of valves to vent the Unit 3 containment was begun on 13 March at approximately 8:41 using air cylinders and an engine generator. Several attempts were made to open the valves and at 9:20 successful venting was confirmed by the decrease in dry well pressure; however, due to the leakage of air, an engine driven air compressor was finally used to provide the required air pressure. At 11:01 on 14 March, a hydrogen explosion occurred in the Unit 3 reactor building resulting in substantial damage. At approximately 6:00 on 15 March, an explosion occurred in the Unit 4 reactor building. Since the spent fuel in the Unit 4 spent fuel pool appears to have been covered with water precluding the generation of hydrogen, the source of flammable gas is unclear. A potential source is hydrogen in the Unit 4 reactor building backflowing from the Unit 3 standby gas system lines through the vent lines of Unit 4. Units 3 and 4 share a common header that vents to the exhaust stack. This is not confirmed. Plans have been made to inert the Unit 3 containment with nitrogen in the future.

MAAP Calculations of the Unit 1-3 Core Degradation Sequence

TEPCO has performed a simulation of the accident using the Modular Accident Analysis Programme (MAAP) code. The information below is only an estimate of the core behaviour.

Based on the calculation, assuming an estimated injection rate, the top of active fuel (TAF) was reached in Unit 1 about three hours after the plant trip. The core was completely uncovered two hours later. Core damage is calculated to have begun four hours after the trip and a majority of the fuel in the central region of the core was melted at 5.3 hours after the trip. At 14.3 hours after the trip, the core was completely damaged with a central molten pool and at 15 hours after the trip, all fuel had slumped to the bottom of the vessel. Although the calculation shows that the RPV is severely damaged, measured data show much cooler temperatures. Due to the uncertainty in the instrumentation at Dai-ichi, the state of the vessel is unknown.

The calculation of the accident progression of Unit 2 is based on an assumed seawater injection rate such that the reactor water level was maintained at about the midpoint of the active fuel as measured by the instrumentation available during the event. The calculation shows that when the RCIC system was available, the water level was maintained well above the TAF. Once RCIC was lost and the system was depressurized the water level dropped to the bottom of active fuel (BAF) about 76 hours after the trip. Seawater injection was initiated and according to the instrumentation, the water level remained at the midpoint of the active fuel region, leading to a rapid increase in core temperature, reaching the melting point. A molten pool existed in the central region of the core with melted fuel surrounding it at 87 hours after the trip. The molten pool was shown to grow larger by 96 hours and then begin to cool at 120 hours. At one week after the trip, there was a small molten pool surrounded by melted fuel. Due to the uncertainties in instrumentation which gave information about the selection of seawater injection rate, another calculation was performed using a reduced rate. This model shows that the fuel has slumped and in turn the RPV is extremely damaged at 109 hours after the trip. Although the calculation shows that the RPV is severely damaged, measured data show much cooler temperatures. Due to the uncertainty in the instrumentation at Dai-ichi, the state of the vessel is unknown.

The calculation of the accident progression at Unit 3 is based on an assumed seawater injection rate such that the reactor water level was maintained at about 3 m below the TAF, as measured by the instrumentation available during the event. The calculation shows that the core was covered until the RCIC and HPCI systems failed. Once seawater injection was initiated and the water level stayed at around 3 m below the TAF, the temperature of the core increased quickly, reaching the melting point. The extent of fuel melting is less than that of Unit 1. This is presumed to be because the time between failure of the RCIC and start of the HPCI system was smaller than the time of no injection in Unit 1. At 64 hours after the trip, a molten pool smaller than Unit 1 was surrounded by melted fuel, and a week after the scram the molten pool had cooled somewhat. No slumping of the fuel to the bottom of the RPV was predicted. Due to uncertainties in instrumentation which gave information about the selection of the seawater injection rate used in the calculation, another calculation was performed using a reduced injection rate. This case predicts that slumping of the fuel occurs at 62 hours after the scram. Although the calculation of this scenario shows that the RPV is severely damaged, measured data show much cooler temperatures. Due to the uncertainty in the instrumentation at Dai-ichi, the state of the vessel is unknown.

Response of Units 5 – 6 and Site Spent Fuel Storage

Units 5 and 6 are located a distance from Units 1–4, and are at a higher elevation than Units 1–4. They suffered less damage than Units 1–4, although the damage was still severe. As a result of the earthquake all off-site power was lost. As in Units 1–4, the seawater ultimate heat sink was lost as a result of the tsunami, and in Unit 5, all EDGs were lost due to flooding. One air cooled EDG was available at Unit 6 because the air intake louvers were located above the tsunami inundation height. Units 5 and 6 had been shutdown since January 2011 and August 2010, respectively, and the fuel had been reloaded into the core recently, awaiting startup. Though decay heat was much lower than the operating plants, cooling the fuel in the cores was necessary and action was taken to restore the seawater cooling system.

On 12 March, measures were successful to provide AC power to important components of Unit 5 using the Unit 6 EDG. On 13 March, the MUWC system was used to inject coolant into the core, and steam was discharged through the SRVs to the suppression pool. Due to the low decay heat of the fuel, venting of the containment was not necessary. On 19 March, an alternate cooling path to cool the RHR system was established. The RHR pump was powered from the Unit 6 EDG. A temporary pump provided sea water to the RHR heat exchangers using an engine-generator to provide AC power. On 20 March, the core was cooled to cold shutdown levels. Plans are underway to provide more heat removal capacity.

ATTACHMENT 46

Energy Research, Inc., ERI/NRC 02-202, "Accident Source Terms for Light-Water Nuclear Power Plants: High Burnup and Mixed Oxide Fuels" (Nov. 2002)

ERI/NRC 02-202

ACCIDENT SOURCE TERMS FOR LIGHT-WATER NUCLEAR POWER PLANTS: HIGH BURNUP AND MIXED OXIDE FUELS

Draft Report: June 2002
Final Report: November 2002

Energy Research, Inc.
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Work Performed Under the Auspices of the
United States Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, D.C. 20555

ERI/NRC 02-202

ACCIDENT SOURCE TERMS FOR LIGHT-WATER NUCLEAR POWER PLANTS: HIGH BURNUP AND MIXED OXIDE FUELS

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**Work performed under the auspices of
United States Nuclear Regulatory Commission
Washington, D.C.
Under Contract Number NRC-04-97-040**

2. BASIS FOR REVISED (NUREG-1465) SOURCE TERM

In 1995, the U.S. Nuclear Regulatory Commission published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" [1], which defined a revised accident source term for regulatory application. NUREG-1465 utilized current technical knowledge and understanding of LWR severe accident phenomenology to present, for regulatory purposes, a more realistic portrayal of the radionuclides present in the containment from a postulated severe accident. NUREG-1465 presents a representative accident source term for a Boiling Water Reactor (BWR) and for a Pressurized Water Reactor (PWR). These source terms are characterized by the composition and magnitude of fission product release into containment, the timing of the release into containment, and the physical and chemical forms in containment.

This chapter provides a summary of the technical basis for the NUREG -1465 accident source term. A brief qualitative discussion of the phenomenology of fission product release and transport behavior during the progression of severe accidents is presented and the technical basis for characterizing the revised accident source term parameters (composition, magnitude, timing and physical and chemical forms) is described.

2.1 Progression of Severe Accident Sequences and Release Phases

Radiological releases into containment under severe accident conditions can be generally categorized in terms of phenomenological phases associated with the degree of core damage and degradation, reactor pressure vessel integrity, and, as applicable, attack upon concrete below the reactor cavity by molten core materials. The general phases, or progression, of a severe LWR accident are shown in Figure 2.1.

Initially there is a release of coolant activity associated with a break or leak in the reactor coolant system (RCS). The radiological releases during the coolant activity release phase are negligible in comparison to the releases during the subsequent release phases. Assuming that the coolant loss cannot be accommodated by the reactor coolant makeup systems, or the emergency core cooling systems, fuel cladding failure would occur. Upon failure of the cladding, a small quantity of fission products that resides in the gap between fuel pellets and the fuel cladding would be released. This release, which is termed the gap release, would consist mostly of the volatile nuclides, particularly noble gases, iodine, and cesium.

As the accident progresses, core degradation begins, resulting in loss of fuel geometry accompanied by melting and relocation of core materials to the bottom of the reactor pressure vessel. Due to temperature variation within the core (impacted by factor such as power density, fuel loading pattern, etc.) the core degradation and subsequent melting and relocation of the core would occur on a region-by-region basis (heterogeneously). Thus, the total release of any radionuclide would occur over a period of time. During this period, the early in-vessel phase, significant quantities of the volatile nuclides in the core inventory as well as small fractions of the less volatile nuclides are released into containment. The fission products and other materials, which are released from the fuel,

ATTACHMENT 47

Excerpt NUREG-1503, "Final Safety Evaluation Report Related to Certification of the ABWR Standard Design," Vol. 1 (July 1994)

NUREG-1503
Vol. 1

Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design

Main Report

Manuscript Completed: July 1994
Date Published: July 1994

Associate Directorate for Advanced Reactors and License Renewal
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

rates when subjected to CCI. This concrete is designed to protect the containment liner in the event that CCI occurs. Basaltic concrete was chosen as it reduces the production of noncondensible gases when compared with other forms of concrete, such as limestone and limestone-common sand concrete. Reduced noncondensible gas and water vapor production from basaltic concrete has been observed during testing (NUREG/CR-5564 and NUREG/CR-5423). However, basaltic concrete does have a lower melting point and quicker ablation rate when compared with limestone and limestone-common sand concrete.

Using the observations discussed above, based on engineering judgment, the staff concludes that the 1.5 m layer of basaltic concrete meets the criteria specified in SECY-93-087 relating to protecting the containment liner and provides sufficient protection for the containment liner.

19.2.3.3.2.1.5 Reactor Pressure Vessel Pedestal

The basaltic concrete discussed above protects the containment liner from core-concrete attack in the axial direction. Core-concrete attack in the radial direction could affect the RPV pedestal. The cylindrical RPV pedestal is formed from two concentric steel rings interspaced with internal stiffeners and filled with concrete. The RPV pedestal is rigidly connected to the diaphragm floor and separates the lower drywell from the wetwell while supporting the loads from the RPV and the diaphragm floor. Built into the RPV pedestal is the drywell-to-wetwell connecting vent system that directs steam from the lower drywell to the suppression pool and upper drywell.

The inner diameter of the RPV pedestal is the outer boundary of the lower drywell. As such, the pedestal is the radial barrier to the horizontal flow of corium. If corium contacts the RPV pedestal, the inner steel cylinder would be attacked and the concrete fill would be subject to ablation. Unabated ablation could lead to failure of the pedestal and subsequent collapse of the RPV and diaphragm floor leading to gross containment failure.

The width of the RPV pedestal is 1.7 m (5.6 ft). The steel rings and internal stiffeners provide the design strength for the RPV pedestal, while the concrete strength is not considered. In Section 19EC of the ABWR SSAR, GE presents the results of an analysis that indicate that only the steel outer shell and 15 cm (6 in.) of internal stiffeners are required to maintain RPV pedestal loads below 90 percent of yield strength.

The staff performed an estimate of the stresses in the RPV pedestal based on the methodology in "Formulas for Stress and Strain," by R. J. Roark and W. Young, McGraw Hill, 1982. Based on these approximate calculations, the staff concludes that adequate margin exists to the yield strength of the RPV pedestal following 1.5 m (5 ft) of radial ablation and that the RPV pedestal is thick enough to withstand the effects of radial ablation resulting from CCI. These attributes meet the criteria specified in SECY-93-087 for protecting structural members with concrete.

19.2.3.3.2.1.6 Containment Overpressure Protection System

The COPS passively relieves containment pressurization before containment pressure reaches ASME Service Level C limits. This system provides for a controlled release through a containment vent pathway with fission product scrubbing provided by the suppression pool. With respect to CCI, the COPS prevents catastrophic overpressurization failure of the containment for severe accident sequences involving prolonged periods of CCI. The COPS ensures that containment pressurization resulting from CCI does not exceed the ASME Service Level C limit of 666.9 kPa gage (97 psig), as the actuation setpoint is 617.8 kPa gage (90 psig).

19.2.3.3.2.2 Analyses

In SECY-93-087, the staff concluded that the evolutionary light water reactors should ensure that the best estimate environmental conditions (pressure and temperature) resulting from core-concrete interactions do not exceed service Level C for steel containments or factored load category for concrete containments, for approximately 24 hours. In addition, designers should ensure that the containment capability has a margin to accommodate uncertainties in the environmental conditions from core-concrete interactions.

The staff concluded that twenty-four hours was an appropriate time period based on sufficient time to allow for decay of fission products, operator intervention, utilization of accident management strategies, fission product deposition in the containment through natural mechanisms, and offsite protective measures. It was developed as a guideline and not a strict criterion in recognition of the uncertainties in severe accident progression and phenomenology.

19.2.3.3.2.2.1 GE Analyses

In Section 19E.2 of the ABWR SSAR, GE provided the results of its deterministic evaluation for several specific accident challenges to evaluate the containments performance. To perform this evaluation, GE used the MAAP3.0B code modified to model the configuration of the ABWR. The new version of the code is referred to as MAAP-ABWR.

Using the ABWR probabilistic safety assessment, GE considered accident classes representing the largest frequencies in selecting the accident sequences to be studied. Eight accident sequences were selected for analysis using MAAP-ABWR. These accident sequences include loss of core cooling with the reactor vessel failing at low and high pressure, SBO, loss of containment heat removal, large break loss-of-coolant accident, and ATWS at low and high pressure and ATWS concurrent with an SBO. For each accident sequence, several mitigating systems could be used to prevent or reduce the release of fission products into the environment. These mitigating systems include in-vessel recovery, passive flooder system, ACIWA, containment heat removal, and containment sprays.

The results of the analyses for each accident sequence are presented in summary form in Table 19E.2-16 of the ABWR SSAR. These analyses generally indicate core debris coolability and little, if any, CCI. The time-to-release of fission products ranges from 8.6 to 50 hours from the start of the transient with the most likely fission product release location through the COPS. The COPS prevents the containment pressure from reaching the ASME Service Level C limit. However, for some sequences, the time to COPS actuation is less than 24 hours. For example, the accident sequence resulting in a release time of 8.6 hours is of extremely low probability involving an SBO with failure of the combustible gas turbine concurrent with an ATWS in which all reactivity control fails. However, if credit is given to operation of the ACIWA in the containment spray mode, the time-to-release of fission products increases to 26.4 hours.

A benchmark of the containment's passive pressure capability is its ability to accommodate the loss of containment heat removal sequence analyzed by GE in section 19E.2.2.4 of the ABWR SSAR. This analysis assumes that reactor vessel injection is maintained with all the decay heat being transferred to the suppression pool. Core damage does not occur. This analysis indicates that the COPS would actuate in approximately 21.7 hours. COPS actuation results from saturation of the suppression pool pressurizing the containment. This sequence indicates that the time to COPS actuation, even without the added pressurization and energy sources from severe accidents, cannot be extended much beyond 20 hours in the absence of active decay heat removal.

With the addition of noncondensible gases from CCI and heat from the exothermic metal-water reactions during a severe accident, the time to COPS actuation will be less. This is an important point in that COPS actuation before 24 hours cannot be prevented unless additional heat capacity is added to the containment or a containment heat removal system is recovered. The ACIWA system, as discussed above in Section 19.2.3.3.2.1.3, can provide additional heat capacity to prolong the time to COPS actuation. Based on GE's analysis provided in Table 19E.2-16 of the ABWR SSAR, the time to COPS actuation is delayed by at least 10 hours for cases in which additional water is added to the containment by the ACIWA, when compared with the same sequence in which only the LDF system actuates to cool the core debris. The ACIWA is crucial to delaying the time to COPS actuation.

Section 19EC presents the results of an uncertainty analyses performed by GE using MAAP-ABWR to investigate the uncertainties associated with debris coolability. These analyses evaluated the impact of parameters such as the amount of core debris, debris-to-water heat transfer, amount of steel in the debris, delayed flooding of the lower drywell, and use of the ACIWA system on CCI, containment pressurization, COPS actuation, and fission product release.

As discussed in Section 19.2.3.3.2.1.4 above, the ABWR will have a 1.5 m (4.9 ft) layer of basaltic concrete above the containment liner. This concrete layer is designed to protect the containment liner from being breached in the event that significant CCI occurs. In Section 19EC using the MAAP-ABWR code, GE provided the results of an uncertainty analyses that calculated the extent of axial ablation. The results, provided in Table 19ED.5-2 of the ABWR SSAR, indicate that axial ablation will not exceed 1 m (3.3 ft) in a 24-hour period.

As discussed in Section 19.2.3.3.2.1.5 above, GE indicated that the distance the molten corium must ablate in the radial direction is 1.55 m (5.1 ft) before the minimum wall thickness of the pedestal is reached. In Section 19EC using the MAAP-ABWR code, GE provided the results of an uncertainty analyses that calculated the extent of radial ablation by multiplying the axial ablation depth by 1/5. GE selected the 1/5 value based on the results of previous CCI experiments. This multiplying factor was necessary, as MAAP assumes that radial and axial penetration are identical. The results, provided in Table 19ED.5-2 of the ABWR SSAR, indicate that radial ablation does not represent a significant threat to the containment.

19.2.3.3.2.2.2 Staff Analyses

The staff analyzed in-house the response of the ABWR using the MELCOR code. In addition, the staff's contractor Sandia National Laboratories (SNL) performed additional analyses using the MELCOR code. The results of the SNL evaluation were sent to GE and placed on the docket. The MELCOR results generally reproduced the event sequences predicted by MAAP, albeit usually with timing shifts. These timing shifts did not affect the safety insights for the containment analyses.

19.2.3.3.2.2.3 Conclusions

The staff did not rely on any one specific sequence or scenario performed by GE using the MAAP-ABWR code nor by the staff's contractor (Sandia National Laboratories) in determining whether the ABWR met the criterion in SECY-93-087 for ensuring that containment conditions do not exceed Service Level C for approximately 24 hours from CCI. Rather, the staff evaluated the range of results provided by these codes, with due consideration of the uncertainties inherent within them, and the capability of the design to extend the time period to COPS actuation through intervention. The ACIWA is fundamental to prolonging the period to COPS actuation. Once COPS is actuated, containment pressurization is relieved through a controlled pathway that takes advantage of scrubbing by the suppression pool. The staff recognizes that there are sequences in which COPS actuation in under 24 hours is required to maintain containment stresses below ASME Service Level C limits.

The staff concludes that the ABWR design meets the criterion when use of the mitigation systems incorporated into the design is factored in, such as the LDF and ACIWA system.

19.2.3.3.2.3 Basis for Acceptability

The ABWR meets the criteria of SECY-93-087 and the staff's proposed applicable regulation for core debris coolability through (1) providing a lower drywell unobstructed floor area greater than 79 m^2 (850 ft^2) to enhance debris spreading, (2) providing an LDF system and ACIWA system to flood the lower drywell, (3) providing a 1.5 m (4.92 ft) layer of basaltic concrete to protect the containment liner, (4) providing a thick reactor vessel pedestal, and (5) providing a COPS. Containment conditions resulting from CCI can be maintained below Service Level C for approximately 24 hours, through incorporation of the above-listed design features.

19.2.3.3.3 High-Pressure Core Melt Ejection

High-pressure core melt ejection (HPME) and subsequent DCH are severe accident phenomena that could lead to early containment failure resulting in large radioactive releases into the environment. HPME is the ejection of core debris from the reactor vessel at a high pressure. DCH is the sudden heatup and pressurization of the containment resulting from the fragmentation and dispersal of core debris within the containment atmosphere.

19.2.3.3.3.1 Preventive and/or Mitigative Features

In SECY-90-016, Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements, the staff concluded that evolutionary LWR designs should include a depressurization system and cavity design features to contain ejected core debris. In its June 26, 1990, SRM, the Commission approved the staff's position that evolutionary LWR designs include a depressurization system and cavity design to contain core debris. In addition, the Commission stated that the cavity design, as a mitigating feature, should not unduly interfere with operations including refueling, maintenance, or surveillance activities.

In SECY-93-087, Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs, the staff recommended that the Commission approve the general criteria that the evolutionary LWR designs provide a reliable depressurization system and cavity design features to decrease the amount of ejected core debris that reaches the upper containment. In its July 21, 1993, SRM, the Commission approved the staff's position.

Based on engineering judgment, the staff believes that examples of cavity design features that will decrease the amount of ejected core debris that reaches the upper containment include ledges or walls that would deflect core debris and an indirect path from the lower drywell to the upper containment. The staff position within SECY-93-087 evolved from the staff position in SECY-90-016 and forms the basis for the staff's review and evaluation.

Therefore, the staff's proposed applicable regulation for high-pressure core melt ejection is as follows:

> The standard design must provide a reliable means to depressurize the reactor coolant system and cavity design features to reduce the amount of ejected core debris that may reach the upper containment so that the potential for and effects of interactions with molten core ejected under high pressure are reduced.

The ABWR has an ADS that is discussed in Sections 5.2.2, 6.3, 7.3, and 19D.6.2.5 of the SSAR. The staff's evaluation of the ADS is provided in Sections 6.3 and 7.3.1.2 of this report. The ADS is a safety grade

ATTACHMENT 48

Excerpt from NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," Vol. 1 (Sept. 2004)

19.1.3.3 Results and Insights from the Level 3 PRA (Offsite Consequences)

In the updated AP1000 PRA, the endstates of the CETs were grouped into six individual release categories. For each release category, the timing, energy, isotopic content, and magnitude of release were established based on plant-specific, thermal-hydraulic calculations using the MAAP code. The NRC-developed MACCS2 code, Version 1.12, was then used to calculate offsite consequences for each of the release categories, specifically, the effective dose equivalent (EDE) whole-body dose complementary cumulative distribution function (CCDF) at 0.80 km (0.5 mile) from the reactor site, and the total person-rem exposure over a 80.4 km (50 mile) radius from the plant. These analyses were supplemented by sensitivity analyses to assess the impact of uncertainties in key parameters. The staff finds this overall approach and the use of the above codes to be consistent with the present state of knowledge regarding severe accident modeling and are, therefore, acceptable.

The following sections present the results and insights from the Level 3 portion of the PRA. This includes the estimated probability of exceeding selected dose criteria, a breakdown of the total risk in terms of important release classes, and a summary of the risk-significant insights from the Level 3 PRA and supporting sensitivity analyses.

19.1.3.3.1 Risk Results for AP1000

Based on the updated PRA, the probability of exceeding a whole-body dose of 0.25 Sv (25 rem) at 0.8 km (0.5 mile) is about 1.9E-8/yr for internal events. This value is about a factor of 50 lower than the Commission's LRF goal of 1E-6/yr and is, therefore, acceptable. The design also meets the public safety requirement goal established by EPRI in the ALWR URD (1E-6 probability of exceeding a dose of 0.25 Sv (25 rem) at a distance of 0.8 km (0.5 mile)). It should be noted, however, that the EPRI goal applies to both internal and external events, and that the results for AP1000 do not include the contribution from seismic and fire events.

Based on the Level 3 PRA, the estimated total risk to the public for AP1000 is quite small. The applicant's analysis indicates a total dose of about 0.05 person-rem/yr, based on the use of population and weather data developed by EPRI to bound 80 percent of the reactor sites in the United States (see Revisions 5 and 6 of the URD), and site land use and crop data based on representative data from the Surry site (NUREG/CR-6613). Those site sectors that are ocean were treated as land in this assessment. Offsite risk is very low compared to the current generation of operating plants because of a combination of (1) a very low estimated CDF for AP1000, (2) a low CCFP, and (3) a relatively benign source term associated with the frequency-dominant release category.

19.1.3.3.2 Leading Contributors to Risk from Level 3 PRA

Table 19.1-5 and Figure 19.1-3 of this report present the contribution to risk from each of the release categories. The following observations can be noted:

- Based on Figure 19.1-3, the probability of exceeding 0.25 Sv (25 rem) at the site boundary (0.8 km (0.5 mile)) is essentially flat and close to unity for all release