State-Of-The-Art Reactor Consequence Analyses (SOARCA)

The U.S. Nuclear Regulatory Commission's (NRC's) State-of-the-Art Reactor Consequence Analyses (SOARCA) research project estimates the realistic outcomes of severe nuclear power plant accident scenarios that could release radioactive material into the environment.

The NRC, industry, and international nuclear safety organizations have extensively researched plant response to hypothetical accidents that could damage the core and containment. This research has significantly improved the NRC's ability to analyze and predict how nuclear plant systems and operators will respond to severe accidents.

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

The SOARCA team applied this accumulated research and incorporated plant enhancements into improved, detailed, and integrated computer models. These models are designed to realistically evaluate the plant behavior during severe accidents, the potential public health risk from a hypothetical release of radioactive material, and consider onsite and offsite actions that may minimize or prevent consequences, such as the implementation of security-related mitigating measures and protective actions for the public (e.g. evacuation and sheltering).

This study is an in-depth analysis of two operating nuclear power plants. Following completion of the study, the NRC will consider applications of its results.

What would you like to learn more about?

- Overview of SOARCA
- What is the SOARCA process?
- Frequently Asked Questions
- Related Information
- Contact Us (link does not exist)
- Public Meetings (links do not exist)

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- Peer Review (link does not exist)

Overview of SOARCA

The focus of this project is to determine the realistic consequences for the most likely severe accident scenarios. A severe accident is a type of accident with a remote likelihood, but may challenge safety systems at a higher than designed level, and may lead to severe reactor core damage. Since the realistic modeling of most likely, non-security related, full-power severe accidents is the focus of this project, the results of this project do not represent a complete risk analysis.

The SOARCA project uses modeling techniques to understand how a reactor might behave under severe accident conditions and how a release of radioactive material from the plant would impact the public. The SOARCA project uses extensive information about many facets of reactor accidents, including reactor systems, components, operating history, impacts of emergency procedures, weather conditions, emergency planning, and population data. It uses the computer code called MELCOR to model the severe accident scenarios within the plant, and the computer code called MACCS2 to model offsite consequences.

SOARCA analyzed an example of each major type of operating U.S. nuclear reactors: a boiling-water reactor (BWR) and a pressurized-water reactor (PWR). Since this project is independent of any regulatory action, nuclear power plants are under no obligation to participate. However, Peach Bottom Atomic Power Station in Pennsylvania and the Surry Power Station in Virginia have volunteered and they are the focus of this report (These two sites were also part of earlier accident analyses). SOARCA analyzed one reactor at each site. The NRC's Advisory Committee on Reactor Safeguards and independent, external experts reviewed the methods and results of SOARCA. The NRC will consider extending SOARCA to analyze additional examples of operating U.S. nuclear reactors.

Next: What is the SOARCA process?

What would you like to learn more about?

- What is a boiling water reactor (BWR)?
- What is a pressurized water reactor (PWR)?
- Peach Bottom Atomic Power Station
- Surry Power Station
- Reactors near you

- MELCOR

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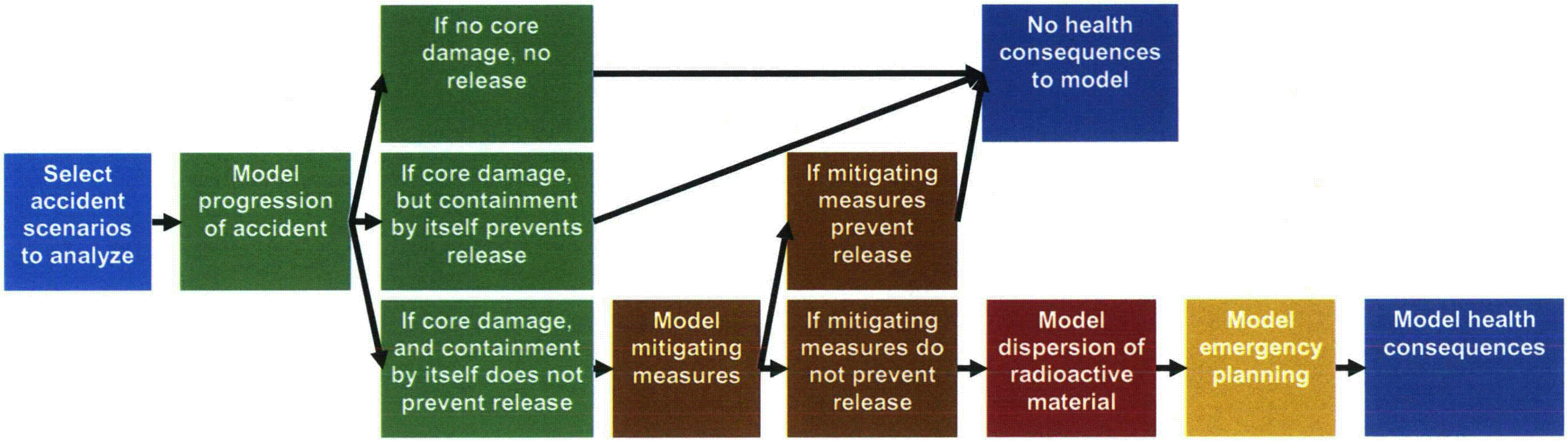
How Is A Severe Accident And Its Consequences Modeled?

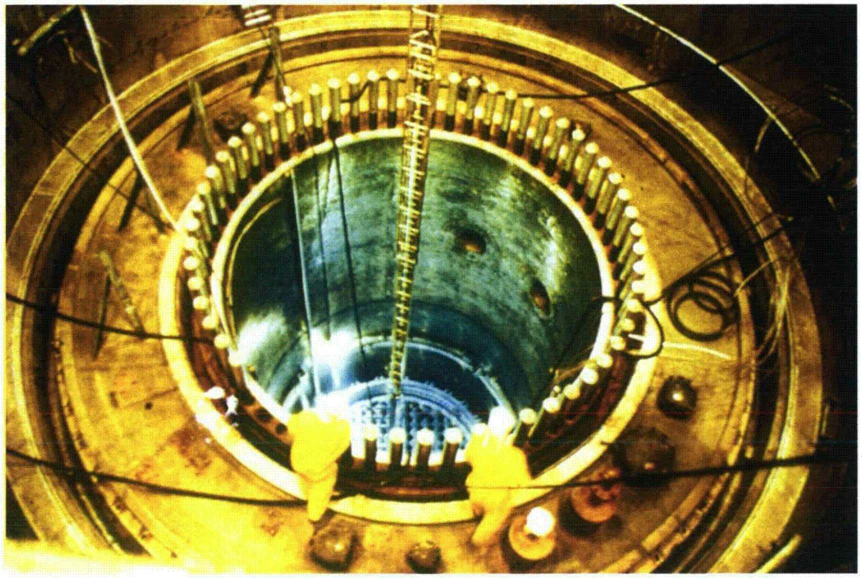


The overall project process consists of six general steps that are illustrated by the colors in the "Flowchart of SOARCA Process" Figure, described very briefly below in an overview, and then described in detail in the following sections. These process descriptions explain the methods and rationale for how the research team proceeded to answer the question what would be the consequences for the most likely severe accidents.

How Were The Scenarios Selected?

The project team modeled accidents scenarios that were most likely to lead to core damage. This step allowed the team to focus our attention and resources on the consequences of most likely severe accidents, and therefore create more realistic, more detailed, and integrated evaluations. Because of the robust, redundant, and diverse safety systems, many things have to fail to reach core damage. Each possible path to core damage is called a sequence. An accident scenario is a group of event sequences that lead to substantial damage of the reactor core. Because of their similarity in how the accident develops, these sequences can be analyzed as one.

How Were The Accidents Modeled?



The project team modeled the best estimate of how each accident scenario would occur. We used data specific for each of the sites' plant design and operation, to provide input to updated models of the major systems of a nuclear power plant and their associated interactions in the severe accident plant behavior code, MELCOR. Since the site has severe accident mitigation equipment, strategies, and procedures that were not in place (or credited) in previous studies, and because the project team the team has updated analytical methods, the team was able to more realistically model how the accidents would progress at each site.

What Are Mitigation Measures?

The NRC requires each site to have emergency operating procedures (EOPs), severe accident management guidelines (SAMGs), and security-related mitigation measures and equipment. Because the mitigating actions may affect the accident progression of a potentially severe accident, these actions are considered in the computer modeling of the accident progression at each site.

How Was the Offsite Release of Radioactive Material Modeled?

After modeling the core damage and containment failure, the project team modeled in the offsite consequence code, MACCS2, how the radioactive material would disperse from the site through the environment and to the population.

How Were the Emergency Response Plans Modeled?

The project team models the emergency plans in MACCS2. For the SOARCA project's purposes of calculating consequences, evacuation of the public in the plant's Emergency Planning Zone (EPZ) is the most evident part of the Emergency Plan. However, the Emergency Plan includes many things, including implementing protective measures, notifying response organizations and the public, etc.

How Were the Health Effects Modeled?

The project team models the consequences from the severe accident MACCS2. The MACCS2 code calculates the exposure to the population, and then uses a dose-response model to determine the consequences of the severe accident in terms of early fatality risk (the risk of an individual dying from radiation sickness in the weeks or months after the event) and latent cancer fatality risk (the risk of an individual contracting fatal cancer due to radiation exposure years after the event).

What Are the Results of SOARCA?

The results of consequence modeling indicate that commercial nuclear power plants are designed and regulated to prevent accidents and to protect the public should an accident occur. In an unlikely situation of both a severe accident and operators unsuccessfully enacting emergency procedures, the consequences to the public are small.

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Previous: Overview of SOARCA

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How Were the Accident Scenarios Selected?

The purpose of SOARCA is to create an improved and realistic evaluation of consequences from severe nuclear accidents. Focusing on select accident scenarios allows us to create more realistic, more detailed, and integrated evaluations of severe accidents.

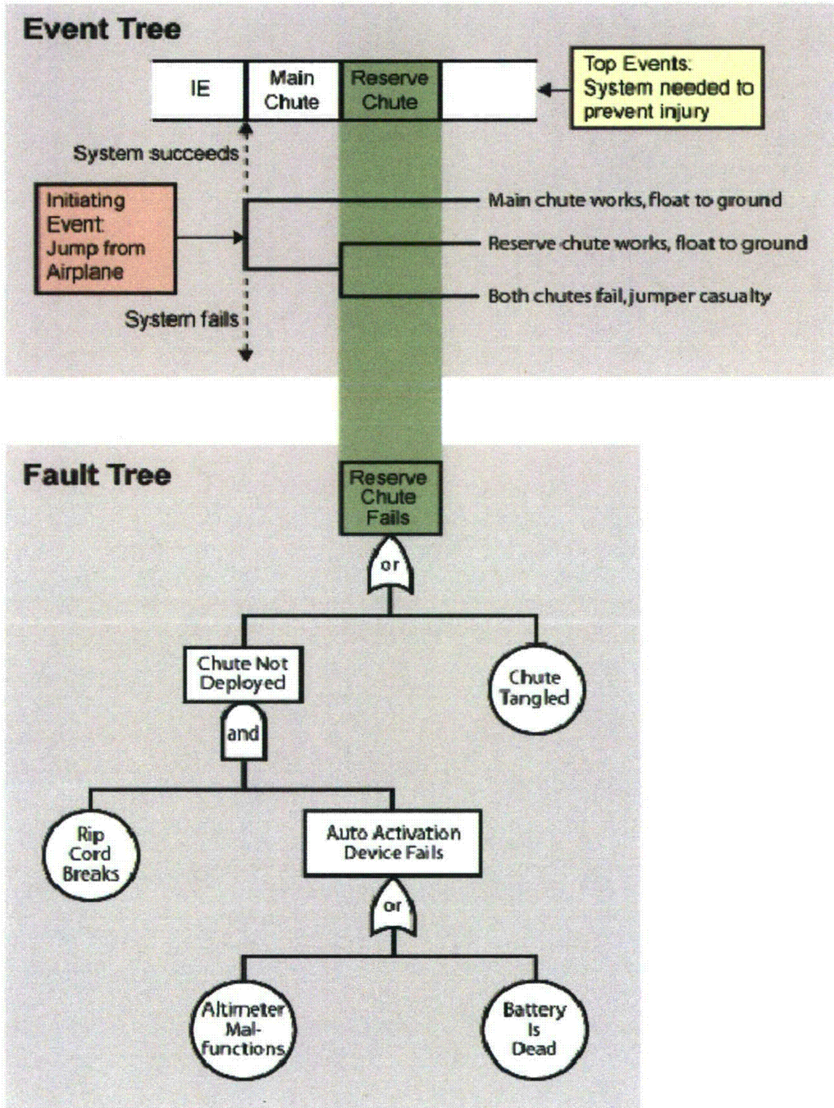
The best scenarios for SOARCA analysis would likely be those with the most risk; however, this is not well known. The NRC has much more information about the likelihood of core damage and accident progression inside the plants than about offsite consequences. Because of this, as a matter of practicality for selecting scenarios, we use core damage frequency (CDF) as a surrogate for risk, in very much the same way risk-informed regulatory decision-making does.

In addition, using CDF to select accident scenarios makes sense because significant release without damaging this first "defense-in-depth" barrier (i.e. the fuel rods in the reactor core) is not possible.

To help identify scenarios with a relatively higher CDF, the staff used the Enhanced Standardized Plant Analysis Risk (SPAR) models. The SPAR models are the NRC's probabilistic risk assessment (PRA) model for quantitatively measuring the likelihood of core damage, and it has specific information about each sites' design, systems, components, and their interdependencies. In addition, the project also considered the licensees' PRAs and previous NRC-sponsored studies.

The staff analyzes accident scenarios with a CDF higher

Sample PRA



than 10^{-6} (i.e. "one-in-a-million"). This one-in-a-million per year CDF selection criterion is an engineering judgment and a value judgment. A CDF selection criterion should be small enough to observe risk-significant consequences, yet large enough that it does not compromise resources in creating realistic, detailed, and integrated evaluations of severe accidents.

This method of selecting scenarios means that we analyze the consequences of the most likely, yet still very remote, severe accident scenarios, and this gives good understanding of the likely consequences if there were a severe accident. However, the NRC does not consider extremely unlikely types of accidents unimportant. For the less likely severe accidents that have a possibility of significantly greater consequence (such as containment bypass or early containment failure scenarios), the staff used a less strict CDF criterion of 10^{-7} (i.e. "one-in-ten-million") to select scenarios for analysis. In addition, the NRC studies the phenomenology of even less likely events to help ensure the health and safety of the public.

The project considered severe accidents from internal events and external events (such as an earthquake) that began during full-power reactor conditions. Security related events, such as a terrorist attack, are analyzed in different, non-public studies.

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Next: How Were the Accidents Modeled?

What would you like to learn more about?

- What is Probabilistic Risk Assessment? (Factsheet: PRA)

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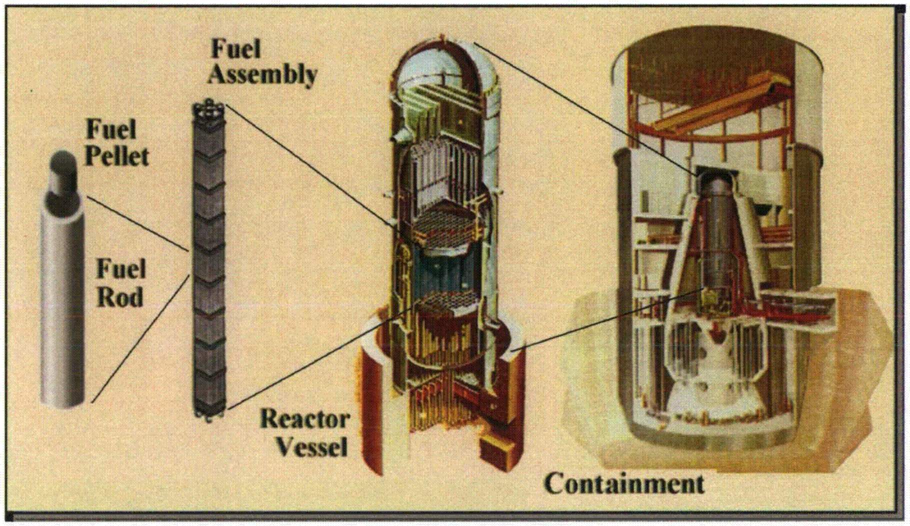
How Were the Accidents Modeled?

After determining what postulated scenarios the project team should analyze, the team determined whether these scenarios could lead to release any radioactive material from the core to the environment through the three "defense-in-depth" barriers. The three "defense-in-depth" barriers are:

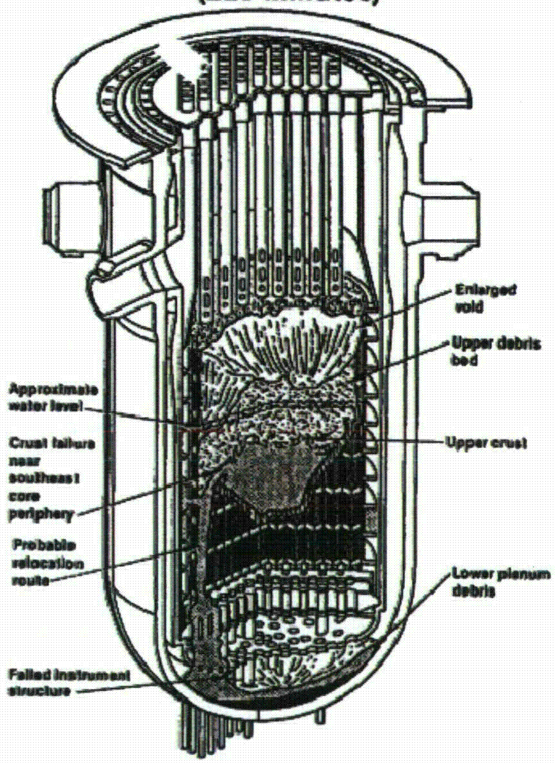
1. The sealed fuel rods in the reactor core
2. The reactor coolant system (reactor pressure vessel and associated components)
3. The containment building



The nuclear fuel pellets are sealed in metal tubes called cladding, and together they are called fuel rods. These fuel rods are contained in the reactor pressure vessel (RPV). The RPV is a thick steel vessel designed for high pressures and is part of the reactor coolant system (RCS). The RCS is inside a special containment building. One type of reactor containment is a large cylinder shaped building made out of reinforced concrete with a steel lining. It is designed to withstand the pressures that might build up inside as steam and gases escape from the reactor during an accident. This containment type is used in the Surry PWR design. Another type of containment, called a pressure suppression containment, has a large water-filled pool to cool the steam and reduce the pressure buildup in the containment. This containment type is used in the Peach Bottom BWR design.

The reactor containment is designed to hold radioactive material that might otherwise be released to the outside environment if the first two "defense-in-depth" barriers fail.

Hypothesized Core Damage Configuration (226 Minutes)



Severe core damage results from accidents that lead to an uncontrolled temperature increase of the reactor core. This heat up may cause the fuel and other core internal structures to melt and relocate. However, all three barriers must fail before a significant release of radioactive material can occur (this includes bypass events, because a bypass of the containment is a type of containment boundary failure). The project team uses the code MELCOR to model the accident progression and plant response for the postulated accident scenarios that may do this.

In the commercial power reactor core melt accident in the United States, the Three Mile Island (TMI) accident in 1979, there was extensive fuel damage. Radioactive gases and contaminated cooling water were released to the containment. Although a little radioactive material was released to the atmosphere by an indirect route, the containment itself performed as designed and kept the vast majority of radioactive material safely inside. The effectiveness of the containment was the major factor in preventing the release of large amounts of radioactive materials to the environment.

In 1986, a much more serious accident occurred at Chernobyl in the former Soviet Union. The Chernobyl accident severely damaged the reactor core, releasing large quantities of radioactive material to the environment. The accident deposited radioactive material in nearby countries, and radioactive material was even detectable at very low levels in the United States. The reactor design (RBMK) is very different from those used in the United States, and the NRC does not allow reactors of that type of reactor design. U.S. reactors have different plant designs, broader shutdown margins, robust containment structures, and operational controls to protect them against the combination of lapses that led to the accident at Chernobyl. The NRC considers this much-analyzed accident not possible in the United States.

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

Long-Term Station Blackout (LTSBO)—In this scenario, the station loses all alternating current power sources but battery backups operate safety systems for about four hours until the batteries are exhausted.

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

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

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

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What would you like to learn more about?

- Three Mile Island Accident
- Chernobyl Accident
- Steam Generator Tube Issues

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What Are the Mitigation Measures?

For the SOARCA project to develop realistic state-of-the-art analyses, the project must include insights into the effectiveness and benefits of mitigation measures at operating reactors. Mitigation measures treated in SOARCA include the site's emergency operating procedures, severe accident management guidelines, and security-related mitigation measures. Security-related mitigation measures refer to additional equipment and strategies required by the NRC following the terrorist attacks of September 11, 2001, to further improve mitigation capability.

Project staff discussed with the nuclear power plant staff the constraints of the postulated accident conditions for each of the scenario the SOARCA project is analyzing. We also discussed with the site staff how the operators would respond to the event. For each scenario, project staff used input from the plant staff to develop timelines of operator actions and equipment lineup or setup times for the implementation of the available mitigation measures. We then used the mitigation measures timelines to develop inputs for MELCOR, the accident progression code.

For each plant accident scenario, the project team models two cases:

> Mitigated Scenario—In the first case, the SOARCA team modeled what would happen if the operators successfully carried out the mitigating actions. The MELCOR calculations included this information to understand how the mitigating actions could affect accident progression. If operators successfully execute these actions, which (by a qualitative evaluation of the known accident constraints, procedures, equipment, and operator training) are a reasonable expectation, consequences to the public will be prevented or minimized.
>
> Unmitigated Scenario—In the second case, the team modeled what would happen if the operators were not successful in carrying out key mitigating actions to prevent the accident from progressing. These cases modeled the sequence of events that lead to core melt, release of radioactive materials, and consequences to the public.

The project considers severe accident mitigation measures as part of the mitigated scenario if:

- the utility has procedures or guidelines for the use of equipment for such a purpose
- the project team can reasonably expect the utility to successfully implement these procedures within the constraints of the accident

NRC inspectors are currently verifying the plant's implementation of security-related mitigation measures.

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How Was the Release of Radioactive Material Modeled?

After core damage and containment failure, radioactive materials are released from plant buildings as aerosol particles in a continuous plume of steam and other gases. The plume of radioactive material disperses into the environment by expanding and moving downwind. This is modeled in the MACCS2 code, which uses site-specific weather, population, and evacuation plans to calculate the exposure to the population from three different pathways:

- cloudshine (exposure from the plume of radioactive aerosols),
- groundshine (exposure from ground contamination), and
- inhalation (exposure from inhaled aerosols).

Because MACCS2 is primarily a probabilistic risk assessment tool, it accounts for the uncertainty in weather that is inherent to an accident that could occur at any point in the future. The results we report is the average risk as a result of the variability in the weather for an average person. SOARCA did not model people to be exposed by eating food on which aerosol particles may have settled, because contaminated food will likely be banned.

Some of the calculated exposure occurs during the early phase of the accident, when the aerosol particles are being released from the plant buildings and while people are evacuating. Some of this exposure occurs in the long term, such as from trace contaminants after land has been decontaminated, or lightly contaminated areas where people never had to evacuate nor relocate. Depending on protective action guidelines and the level of radiation and, these areas may be considered habitable. For the Surry model, SOARCA uses habitability criteria from the U.S. Environmental Protection Agency "Manual of Protective Action Guides for Nuclear Incidents". For the Peach Bottom model, SOARCA uses Pennsylvania-specific habitability criteria.

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How Were the Emergency Response Plans Modeled?

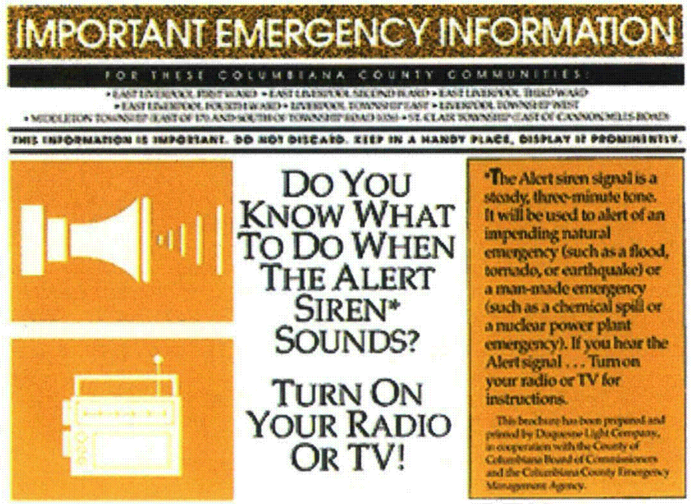
Emergency preparedness (EP) for nuclear power plants are programs, plans, training, exercises, and resources designed to protect public health and safety in the event of a radiological accident. These emergency response programs are developed, tested, and evaluated and are in place as another level of defense to protect the public in the unlikely event of an accident. The NRC requires that each site demonstrate reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. The Emergency Plan regulations, in part include:

- Conducting periodic exercises of emergency response capabilities, and maintaining and correcting these capabilities
- Providing and maintaining adequate emergency facilities and equipment to support the emergency response
- Establishing procedures to notify state and local response organizations and emergency personnel
- Arranging for medical services for contaminated injured individuals
- Developing plans for recovery

In order to evaluate the impact of emergency response actions on public health in a realistic and practical manner, the staff incorporates the response of the public into MACCS2. MACCS2 does this by integrating emergency response plan elements with other aspects of the analysis. By including a simulated evacuation and other protective actions with the atmospheric transport and dispersion model, we can more realistically calculate a predicted exposure to the population.

The project team uses detailed emergency response planning within the 16 km (10 mile) plume exposure pathway Emergency Planning Zone (EPZ), which provides a substantial base for expansion to a larger response area if needed. If the population in areas beyond the EPZ needed to take protective actions, the protective actions would be limited to affected areas based on plume projections. The project team assessed variations of emergency response, which include evacuation and sheltering of population groups outside the 10 mile EPZ to a distance of 20 miles from the plant.

Offsite response organizations (OROs) are expected to act to reduce the risk to the public in the unlikely event of an accident. The project staff obtained site-specific information from OROs to support development of timelines of these protective actions to use in the models. In addition, the licensees provided information, including evacuations time estimates (ETEs). The level of detail in developing these ETEs was significant, including the identification of different evacuation characteristics of the general public and special facility population groups known as "cohorts".



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What would you like to learn more about?

- Emergency Preparedness (Factsheet: Emergency Preparedness at Nuclear Power Plants)

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How Were the Health Effects Modeled?

Radiation is all around us. We tend to think of biological effects of radiation in terms of their effect on living cells. For low levels of radiation exposure, the biological effects are so small they may not be detected. The body has repair mechanisms against damage induced by radiation as well as by chemical carcinogens. Consequently, biological effects of exposure to low levels of radiation on living cells may result in three outcomes:

(1) injured or damaged cells repair themselves, resulting in no residual damage;
(2) cells die, much like millions of body cells do every day, being replaced through normal biological processes; or
(3) cells incorrectly repair themselves resulting in a biophysical change; this represents the first stage of what may eventually become cancer.

In SOARCA, MACCS2 calculates consequences using a dose-response model and the predicted amount of exposure to the population. From this, we calculate the risk of two different types of consequences from the analyzed scenarios:

Early Fatality Risk—The risk of dying from radiation sickness due to an acute dose of radiation.
Latent Cancer Fatality Risk—The risk of dying from a cancer that could occur years after exposure to radiation.

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What would you like to learn more about?

- Radiation (Factsheet: Biological Effects of Radiation; Learn more: Biological Effects of Radiation)

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Results and Conclusions

The SOARCA analysis predicts no early (i.e., caused by radiation sickness) fatalities from the mitigated scenarios, as well as essentially no early fatalities from unmitigated scenarios.

Predicting latent cancer risk is strongly influenced by modeling uncertain, low-dose (<0.1Sv, <10rem) health effects, in part because much of the estimated radiation exposure in the SOARCA scenarios is received after the population returns. SOARCA analyses with any of the four cancer risk estimate models, however, show very small latent cancer fatality risks.

Assuming that any of the analyzed scenarios has occurred, an average individual within 10 miles of a plant has a latent cancer fatality risk as large as 10^{-3} (1 in a thousand), using the linear, no-threshold dose-response model. Other dose-response models yield lower or much lower conditional risks. Keep in mind, however, that successful mitigation measures generally prevent a release.

Including the probability of the severe accident itself reduces an individual's latent cancer fatality risk from any scenario to no more than 10^{-9} (1 in a billion) per reactor-year. This is many orders of magnitude smaller than the U.S. average risk of a cancer fatality or the NRC's Safety Goal. Essentially all of the individual latent cancer fatality risk from the analyzed accidents is from doses less than the U.S. average dose from background radiation.

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Yet to be included:

Key Messages:

- The results of this project indicate that reactor safety has improved over the years as a result of efforts by the commercial nuclear power industry to improve plant design and operation and by NRC to develop improved regulations to enhance safety.
- For the scenarios examined, offsite health effects are usually prevented, because in most cases, we reasonably expect operators have enough time, equipment, and ability to perform the necessary actions. In at least three of the six scenarios, we reasonably expect mitigating actions to prevent core damage.
- For the scenarios examined, our analyses indicate that potential radiation releases would occur several hours later than earlier thought, and they would be substantially smaller; as a result, our best estimate of early fatalities from severe accidents at nuclear power plants would be far less than previously calculated.
- The analyzed scenarios predict that essentially no deaths from radiation exposure will occur within weeks following the accident and long-term cancer fatality risks are very low for the unmitigated scenarios examined.
- The SOARCA individual long-term cancer risk values for the selected scenarios are much smaller than the NRC-established safety goal that "individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health."

LNT and truncation models

http://www.nrc.gov/reading-rm/basic-ref/students/emergency.html
http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/emer-plan-prep.html
http://www.nrc.gov/reading-rm/basic-ref/students.html

GLOSSARY

Core Damage - (an accident leading to) heatup of the reactor core to the point at which severe fuel damage is anticipated -or- uncovery and heatup of the reactor core to the point at which prolonged oxidation and severe fuel damage is anticipated representing the onset of gap release of radionuclides

Source Term - The amount and timing of radioactive or hazardous material released to the environment following an accident.

MACCS2 Code - The computer code used to calculate dispersion of radioactive material to the environment and the population. The MACCS2 code uses a dose-response model so the project team can determine the health consequences of the severe accident in terms of early fatalities (how many people in a population would die in the weeks or months following exposure) and latent cancer risk (the numbers of individuals in a population contracting fatal cancer due to exposure).

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

Radioactive Material

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