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

The State-of-the-Art Reactor Consequence Analyses (SOARCA) project analyzes severe accident consequences to develop a body of knowledge regarding the realistic outcomes of severe reactor accidents. In addition to incorporating the results of more than 25 years of research, the objective of this updated plant analysis is to include the significant plant safety improvements and updates, which have been made by plant owners but were not reflected in earlier assessments by the U.S. Nuclear Regulatory Commission (NRC). In particular, these plant safety improvements include system enhancements, training and emergency procedures, and offsite emergency response. In addition, these improvements include the recent enhancements in connection with security-related events.

The goal of SOARCA is to determine best estimates of the offsite radiological consequences for severe accidents at U.S. operating reactors using a methodology based on state-of-the-art analytical tools, and to present those results using risk communication principles to achieve informed public understanding of the important factors. In particular, those factors include the extent and value of defense-in-depth features of plant design and operation, as well as mitigation strategies. As a result, SOARCA will update ~~outdated~~ analyses, such as NUREG/CR-2239 (SAND81-1549), "Technical Guidance for Siting Criteria Development," dated November 1982.

What would you like to learn more about?

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

Overview of SOARCA

2/3

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 much higher than expected level, and may even lead to severe 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 ~~claim to~~ represent a complete risk analysis.

The first two nuclear power plant (NPP) sites that the project team analyzed are examples of each major type of nuclear reactor in the U.S.: 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 and Surry have volunteered, and they are the focus of the ~~results~~. Peach Bottom is an example of a boiling water reactor and Surry is an example of a pressurized water reactor. As this project develops, the research team may analyze different variations of these two types of reactors.

present phase of the study

How ~~Is~~ are? A Severe Accident And Its Consequences Modeled?

The SOARCA project uses modeling techniques to understand how a reactor might behave under severe accident conditions and how ~~this~~ would impact the public. ~~The models consist of calculations of measurements and estimates about the phenomenon~~. The SOARCA project calculates information about reactor systems, components, operating history, impacts of emergency procedures, weather conditions, emergency planning, and population data, ~~etc.~~ It uses the computer code called MELCOR to model the severe accident scenarios ~~and the~~ computer code called MACCS2 to model offsite consequences.

a release of radioactive material

including models for how the reactor core would degrade and how radioactive material would move through the plant to the environment. These models are based on several decades of research since the last NRC-sponsored study.

The source term results are passed to the

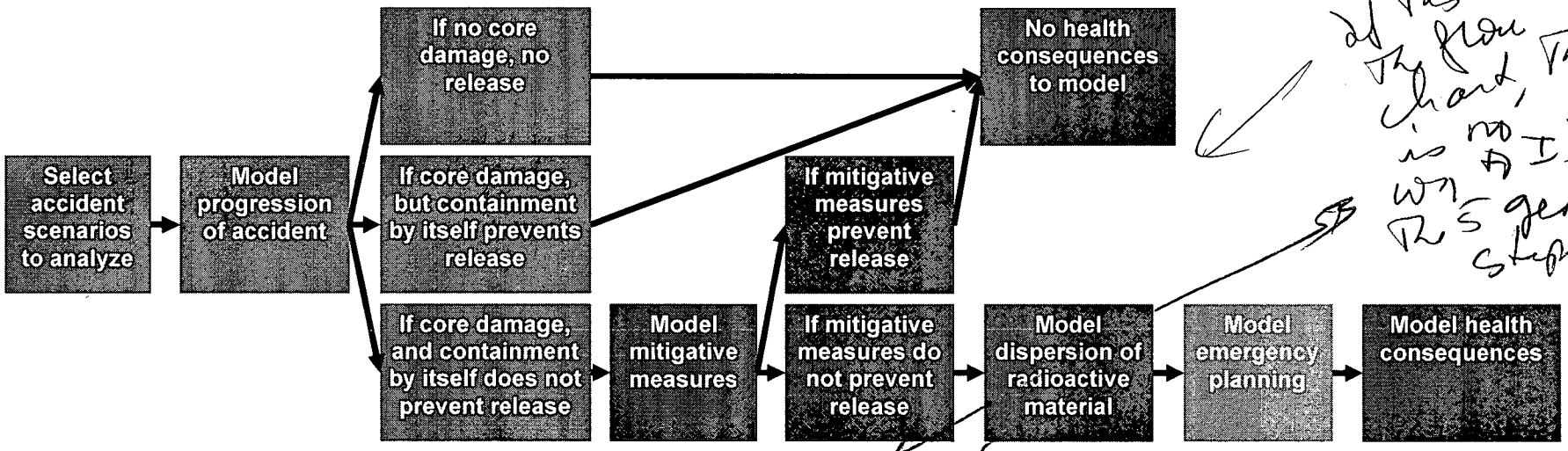
Next: What is the SOARCA process?

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
- MELCOR

Back to main page: SOARCA

What is the SOARCA Process



At this S The flow chart, there is no ID w/ the 5 general steps -

Looks like 6 to me.

Sorry but, maybe here, the this is a quick overview of the 6 steps.

The overall project process consists of five general steps that are illustrated in the "Flowchart of SOARCA Process" Figure and 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 ~~relatively~~ the most likely to ~~have~~ lead to core ~~damage~~ damage of the. This step allowed the team to focus attention and resources on the consequences of most likely severe accidents. 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 ~~similar~~ the event sequences that lead to substantial damage of the reactor core, and can be analyzed as one.

How Were The Accidents Modeled?

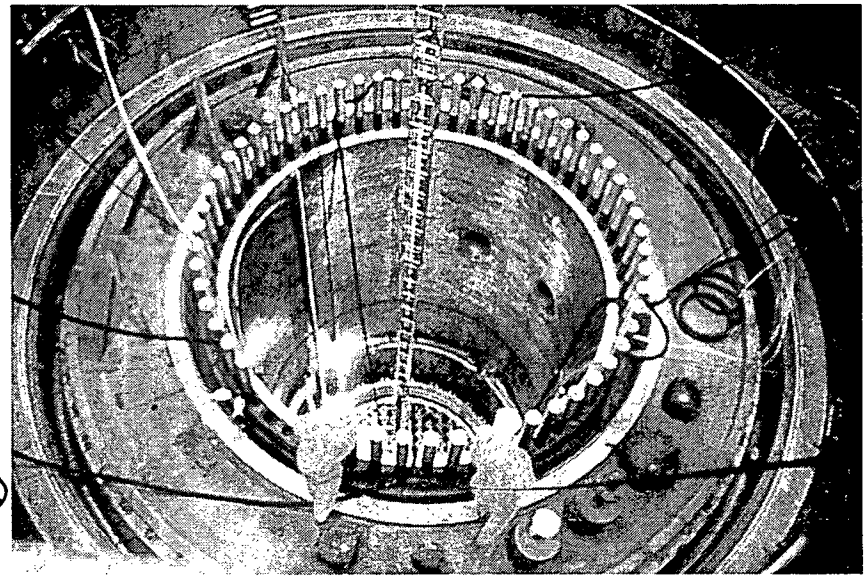
Then we will return to each in more detail

Because of their similarities in how the accident develops,

to provide input to models

begin and progress.

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~~s~~ and operation~~s~~, ~~and updated codes~~ of the major systems of a nuclear power plant and their associated interactions. Since the site has severe accident mitigation equipment, strategies, and procedures that were not in place 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.



error not considered

What Are Mitigation Measures?

(EOPs) management (SAMGs)

The NRC requires each site to have emergency operating procedures, severe accident guidelines, post-9/11 mitigation measures and equipment~~, etc~~. The project team input the mitigation measures into the computer modeling of the progression of the severe accident.

and the operator actions and equipment capabilities of

How Was The Release of Radioactive Material Modeled?

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

How Were The Emergency Response Plans Modeled?

off-site consequence

The project team model~~s~~ed the emergency plans in the computer code ~~for the dispersion of radioactive material~~. For the SOARCA project's purposes of calculating consequence, ~~the~~ evacuation 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.~~

of the public in the plant's Emergency Planning Zone (EPZ)

and relocating the public beyond the EPZ, if necessary.

what does this mean?

How Were The Health Effects Modeled?

The project team model~~s~~ed the consequences from the severe accident in the computer code ~~for the dispersion of~~

off-site consequence

following very shortly after

radioactive material. The MACCS2 code uses a dose-response model to determine the consequences of the severe accident in terms of early fatality risk (the risk to an individual dying from radiation sickness from radiation exposure) and latent cancer fatality risk (the risk to an individual contracting fatal cancer due to radiation exposure).

years after the event

of

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 a low-likelihood situation of a severe accident and operators unsuccessfully enacting emergency procedures, the consequences to the public are very low.

small

Next: How Were The Accident Scenarios Selected?
Previous: Overview of SOARCA

What would you like to learn more about?

Back to main page: SOARCA

How Were The Accident Scenarios Selected?

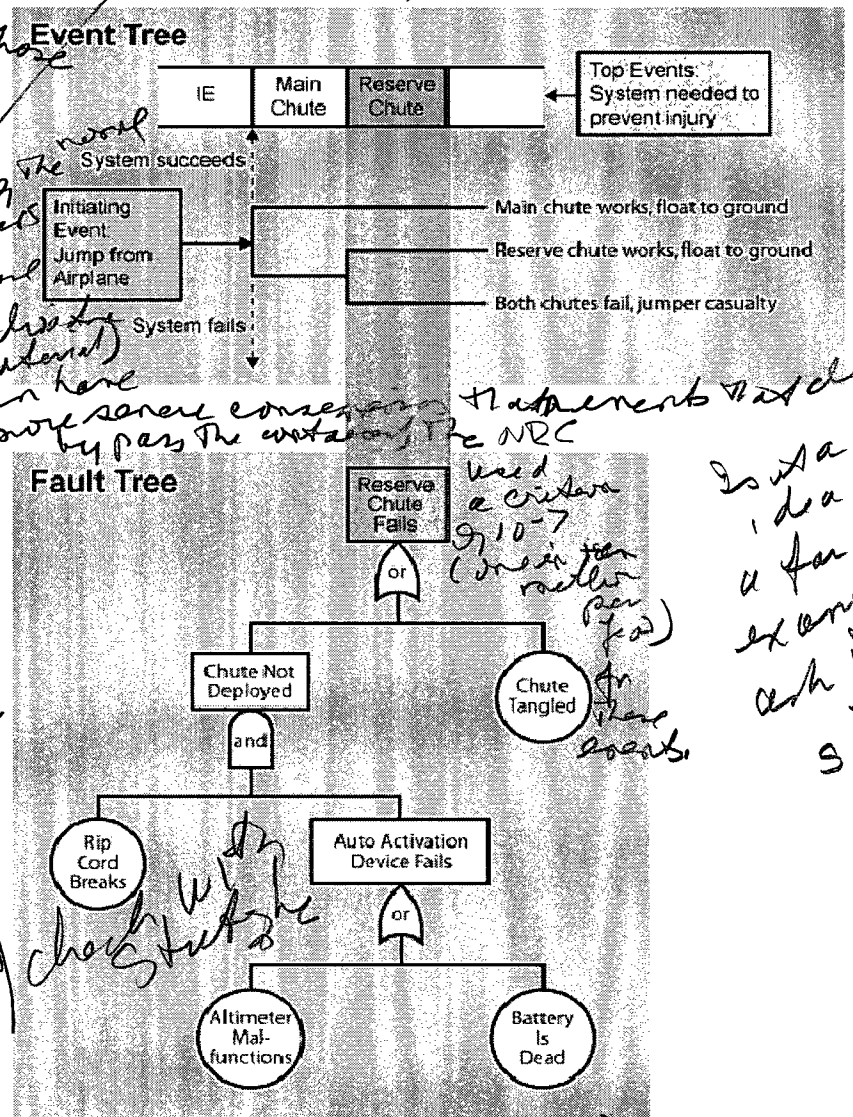
Because a severe accident is an extremely rare possibility, the NRC is designing the project to focus on consequences from the more plausible accident scenarios, as opposed to scenarios with a calculated frequency of less than 10^{-6} ("one-in-a-million") per year. This will give a good understanding of the likely consequences if there was a severe accident.

To help identify scenarios with a relatively higher frequency of core damage, the staff is selecting the accident scenarios using 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 included data about each sites' design, systems, components, etc. The NRC uses an accident's core damage frequency (CDF) as a basis for selecting which accident to analyze for two reasons:

- Significant release without damaging this "defense-in-depth" barrier (i.e. the fuel rods in the reactor core) is not possible
- The NRC's level of CDF understanding is more complete than its understanding of the entire accident progression

The staff analyzes accident scenarios with a CDF higher than 10^{-6} (i.e. "one-in-a-million"). These scenarios encompass more than 90% of the total internal event CDF. (However, because of limitations of the SPAR

Sample PRA



good grief, I hope not!

models, the staff needs to dismiss some of these accident scenarios out of hand with expert judgment). In addition, the project considers accident scenarios that bypass the containment more stringently. Because bypass accidents have an increased associated risk, the staff analyzes these accident scenarios with a CDF greater than 10^{-7} (i.e. "one-in-ten-million").

The project considers severe accidents from internal events and external events (such as an earthquake), but only accidents during at power reactor conditions, and not security related events such as a terrorist attack.

Previous: What is the SOARCA process?
Next: How Were The Accidents Modeled?

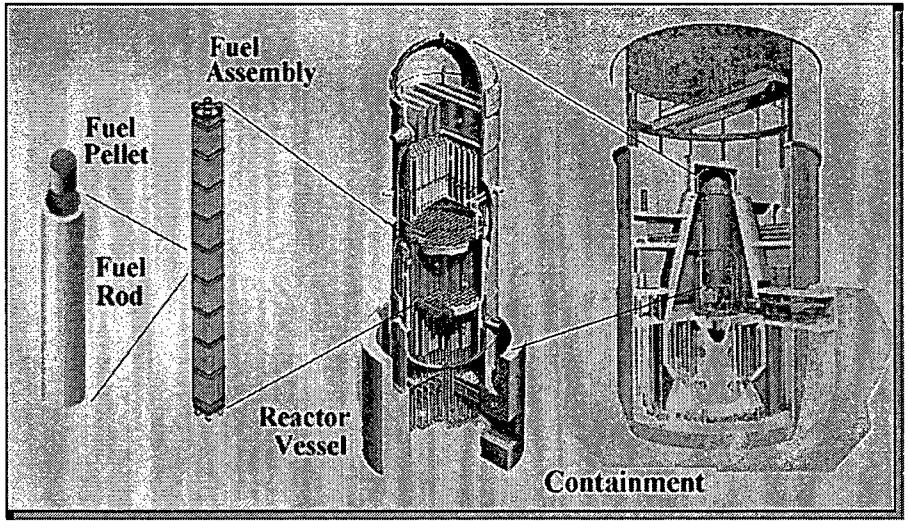
What would you like to learn more about?
What is Probabilistic Risk Assessment? (Factsheet: PRA)

Back to main page: SOARCA

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 & associated components)
3. The containment building



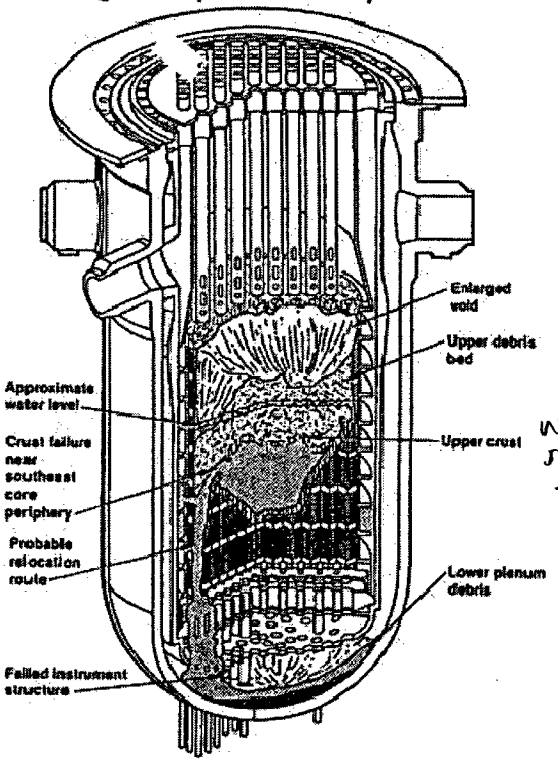
The nuclear fuel is sealed in metal tubes called cladding. These fuel rods are contained in the reactor pressure vessel (RPV). The RPV is a thick steel vessel designed for high pressures and is part of the reactor coolant system (RCS). The RCS is inside a special containment building. One type of reactor 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. 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.

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.

this containment type is part of the Surry pressurized water reactor design.

this containment type is part of the Peach Bottom boiling water reactor design.

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 ~~most~~ fail before a significant release of radioactive material can occur. The project team uses the code "MELCOR" to model the accident progression and plant response for the postulated accident scenarios ~~that may do th~~is. (But one should note that even these postulated accidents scenarios have a statistically-calculated chance of about one-in-a-million years of occurring at a nuclear power plant.)

In the only ~~major~~ commercial power reactor accident in the United States, the Three Mile Island accident in 1979, there was extensive fuel damage. Radioactive gases and contaminated cooling water ~~filled~~ the containment. Although some radioactivity was released to the atmosphere by an indirect route, the containment itself performed as designed and kept the radioactivity safely ~~bottled~~ 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 radioactivity to the environment. The accident deposited radioactive material in nearby countries, and radioactivity 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. The Chernobyl reactor has an inherent instability called a positive void coefficient. ~~In addition, plants in the United States have a better containment structure~~. The NRC considers this ~~much analyzed~~ accident not possible in the United States.

In the SOARCA project, we simulated the following scenarios for both sites (Peach Bottom and Surry):

1. *Long Term Station Blackout (LTSBO)* - In this scenario, the station loses offsite power and both of the emergency diesel generators (EDG). The batteries, however, are available for short term use (about 4-8 hours) to run the safety systems but in the "longer term" there would be no power.

2. *Short Term Station Blackout (STSBO)* – In this scenario, the site loses all power, even the batteries, and therefore all of the safety systems quickly become inactivated in a "shorter term."

We modeled two more scenarios for the Surry site:

1. *Inter-System Loss-of-Coolant Accident (ISLOCA)* - In this scenario, valves which connect the reactor to a low-pressure system fail and cause a rupture that is outside the containment. This accident is called a "bypass event", which means that release occurs through another path without the containment having a chance to prevent the release. It is important to note that this accident scenario is not initiated by an external event.

2. *Thermally-Induced Steam Generator Tube Rupture (TISGTR)* – In this scenario, the reactor is performing under conditions of a Short-Term Station Blackout, but during the accident, extremely hot steam and hydrogen flow out into the steam generator tube. If the tube fails, radioactive material moves through steam line, past containment and exits a relief valve. This accident is a "bypass event". which means that release occurs through a path that the containment cannot prevent the release.

Previous: How Were the Accident Scenarios Selected?
Next: What Are The Mitigation Measures?

What would you like to learn more about?
Three Mile Island Accident
Chernobyl Accident
Steam Generator Tube Issues

Back to main page: SOARCA

(3)

What Are the Mitigation Measures?

For the SOARCA project to develop realistic state-of-the-art analyses of accident progression, radiological release, and offsite radiological health consequences, the project must include insights into the effectiveness and benefits of mitigation measures currently employed at operating reactors.

This includes mitigation measures beyond those treated in current PRA models. Mitigation measures treated in SOARCA include the site's emergency operating procedures (EOPs), severe accident management guidelines (SAMGs), and post-9/11 mitigation measures. Post-9/11 mitigation measures refer to additional equipment and strategies required by the NRC following the terrorist attacks of September 11, 2001, to further improve mitigation capability.

Project staff discussed with the nuclear power plant staff ~~about~~ the constraints of the postulated accident conditions for each of the scenario the SOARCA project is analyzing. Project staff also discussed with the site staff how the operators would respond to the event. For each sequence group, 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. Project staff 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, given the timeline developed with nuclear power plant staff, 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 procedures, then consequences to the public will be prevented or minimized.

Unmitigated Scenario—In the second case, the team modeled what would happen when newly available equipment was not used as additional mitigating actions. These cases modeled the sequence of events that lead to core melt, release of radioactive material, 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 utility has practiced drills using this equipment
- the project team can reasonably expect the utility to successfully implement these procedures within the

check this because, as I remember, P.B. did not have the equipment ordered when SOARCA met with the operators

constraints of the accident

NRC inspectors are currently verifying the plant's implementation and reliability of post-9/11 mitigation measures.

Previous: How Were the Accidents Modeled?
Next: How Was The Release Of Radioactive Material Modeled?

What would you like to learn more about?

Back to main page: SOARCA

How Was The Release of Radioactive Material Modeled?

After modeling the postulated core damage and containment failure, the project team modeled how the radioactive material would disperse from the site to the environment and to the population. This was done with the MACCS2, which is an updated computer code with site-specific meteorology, evacuation plans, and recent population data. MACCS2 models the dispersion of the radioactive material and calculates the human exposure from the cloudshine (exposure from the plume), groundshine (exposure from fallout on the ground), and inhalation (exposure from inhaled aerosols) dose pathways. 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. From this meteorology, we report the mean risk (depending on proximity to the accident) for an average person.

Previous: What Are The Mitigation Measures?
Next: How Were The Emergency Response Plans Modeled?

What would you like to learn more about?

Back to main page: SOARCA

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 defense in depth to respond in the unlikely event of an accident. The NRC requires each site has reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. The Emergency Plan regulations include:

- Conducting periodic exercises of emergency response capabilities, and maintain and correct 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 by all organizations
- Arranging for medical services for contaminated injured individuals
- Developing plans for recovery

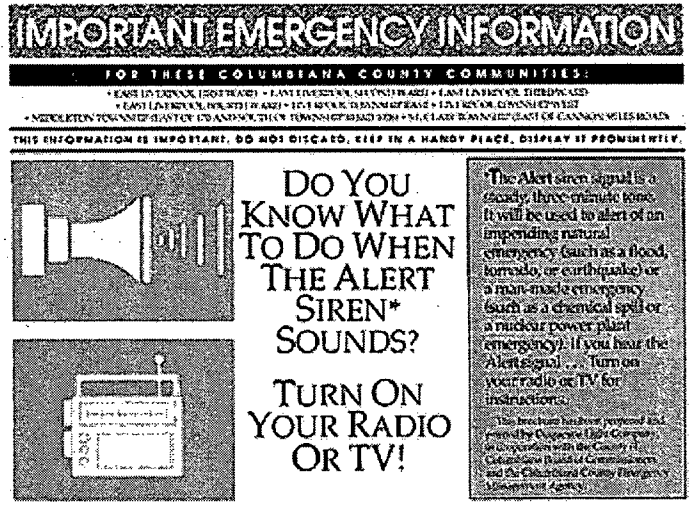
Evaluating the response of the public in order to model emergency response in a realistic and practical manner is an objective of the SOARCA project. In order to support more realistic consequence analyses, the project team uses the code "WinMACCS" to integrate emergency response plan elements and protective actions with the atmospheric transport and dispersion model.

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 NPP.

Offsite response organizations (OROs) will 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

additional information was obtained from the licensee,

protective actions to use in the models, 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"~~.



Previous: How Was The Release Of Radioactive Material Modeled?
Next: How Were The Health Effects Modeled?

What would you like to learn more about?
Emergency Preparedness (Factsheet: EP at Nuclear Power Plants)

Back to main page: SOARCA

(6)

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 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.

In SOARCA, MACCS2 uses a dose-response model to calculate consequences, depending on the amount of exposure to the population. The project team calculates the risk of two different types of consequences from the analyzed scenarios:

Early Fatality Risk—The risk of dying from radiation sickness that occur shortly after exposure to large doses of radiation (usually within a few weeks).

Latent Cancer Fatality Risk—The risk of dying from a cancer that could occur years after exposure to radiation. The project team calculates the radiological dose to the public using the emergency response information and the code for the atmospheric transport and dispersion called MACCS2.

Previous: How Were The Emergency Response Plans Modeled?

What would you like to learn more about?

Radiation (Factsheet: Biological Effects of Radiation)

http://www.nrc.gov/reading-rm/basic-ref/students.html
http://www.nrc.gov/reading-rm/basic-ref/teachers/09.pdf

Back to main page: SOARCA

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

LNT and truncation models

Early Fatalities – fatalities from extreme cases of radiation sickness.

Latent Cancer Fatalities (LCF) - cancer fatalities that occur years following release of a large quantity of radionuclides to the environment

WHAT ARE THE EARLY FATALITIES?

Mitigated Scenarios - In the simulations for each of these scenarios, the operators were able to prevent core damage by using portable equipment put in place following 9/11 and supplemented by the use of existing equipment with new procedures to run under blackout conditions. Since the accidents in these scenarios were effectively mitigated, there is no release of radioactive materials or consequences to the public and environment.

Unmitigated Scenarios - There are no early fatalities for the comparison case scenarios. Even though these scenarios did lead to core damage, the release of fission product occurs after long periods of time which allow for evacuation of the population. Therefore, in these situations, no one is initially exposed to large amounts of radioactive material.

WHAT ARE LATENT CANCER FATALITIES?

Mitigated Scenarios - In the simulations for each of these scenarios, the operators were able to prevent core damage by using portable equipment put in place following 9/11 and supplemented by the use of existing equipment with new procedures to run under blackout conditions. Since the accidents in these scenarios were effectively mitigated, there is no release of radioactive materials or consequences to the public and environment.

Unmitigated Scenarios - In these scenarios, the latent cancer fatalities are modeled to be very small—regardless of which distance interval or calculation model is used. The following tables show the results of the calculations for each plant and scenario.

HOW THE HEALTH CONSEQUENCES MODELED?

This calculation will include ~~both~~ LNT and a 10 mrem threshold models. SOARCA will present results for ~~three~~ distances:

1.) 0-10 miles
2.) 0-50 miles
~~3.) 0-100 miles~~

RESULTS FROM COMPARISON CASES

The ~~base~~ case models demonstrate that reactor designs, operator actions, and regulations can prevent consequences to the public if a severe accident were to occur. However, ~~for the benefit of comparison~~, we ran scenarios that demonstrate the consequences if the operators were unable to execute the emergency procedures. The results are presented below and demonstrate that even in these scenarios the consequences to the public are very low.

GLOSSARY

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

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~~

Early Fatalities - human fatalities that result from doses ~~that are accumulated before effective implementation of offsite emergency response and protective actions (usually within [name timeframe])~~.

Latent Cancer Fatalities - cancer fatalities that occur years following the exposure. ~~release of a large quantity of radionuclides~~ to ~~the environment~~

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 ~~in~~ to within the environment and the exposure of the population. The MACCS2 code uses a dose-response model ~~so the project team can~~ determine the consequences of the severe accident in terms of early fatalities (how many people in a population would die upon initial 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.

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

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

U. S. Nuclear Regulatory Commission. Technical Guidance for Siting Criteria Development. November 1982. NUREG/CR-2239.

- How Were The Scenarios Selected?
- How Were The Accidents Modeled?
- What Are The Mitigation Measures?
- How Was The Release Of Radioactive Material Modeled?
- How Were The Emergency Response Plans Modeled?
- How Were The Consequences Modeled?
- References
- Glossary

Corrective Actions = Mitigative Measures
A model is a tool to predict how a process will perform under a set of conditions.
These calculations are very complex are done by computer programs designed for specific modeling conditions.

Q&A

to have detailed emergency plans about how onsite personnel would work to prevent a release in the case of an accident and, if a release occurred, how offsite personnel would coordinate evacuation and sheltering of surrounding populations. The computer code uses this information to model the evacuation of the public in the case of a severe accident.

-
- Assigning responsibilities to organizations within the EPZ
- Providing prompt communications among principal response organizations to emergency personnel and to the public
-
- Arranging for requesting and effectively using assistance resources
- Developing a range of protective actions for the plume exposure pathway EPZ for emergency workers and the public
-

- Establishing means for controlling radiological exposures for emergency workers