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REPORT BY THE U.S.

General Accounting Office

Response To Specific Questions On The Indian Point Probabilistic Safety Study

This report describes and summarizes how probabilistic risk assessment (PRA) is applied to nuclear powerplants, the methodology and results of a PRA on the Indian Point plants near New York City, and peer review comments on that study.

PRAs attempt to examine complex systems, such as nuclear powerplants, and identify their potential safety, environmental, and economic risks. GAO found that while the Indian Point PRA may represent the state of the art in risk assessment, it suffers from the same limitations as all PRAs: uncertainties and inconsistencies in data, models, assumptions, and methods. Although the study identified the dominant contributors to risk, it did not identify the precise level of risk from operating the Indian Point nuclear powerplants

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UNITED STATES GENERAL ACCOUNTING OFFICE
WASHINGTON, D.C. 20548

RESOURCES, COMMUNITY, AND ECONOMIC DEVELOPMENT DIVISION

B-211642

The Honorable Richard L. Ottinger
Chairman, Subcommittee on Energy
Conservation and Power
Committee on Energy and Commerce
House of Representatives

Dear Mr. Chairman:

In your August 20, 1982, letter you asked that we review the reliance placed on probabilistic risk assessment (PRA) techniques by the Nuclear Regulatory Commission (NRC), with particular emphasis on the safety assessments performed at the Indian Point nuclear powerplants located close to New York City. Specifically, you asked that our review focus on the following questions:

--What is the current state of the art regarding PRA?

--To what extent has NRC incorporated PRA into the regulatory process and does this appear reasonable considering the staff's experience and training?

--What are the problems and potential disadvantages associated with the use of PRA and has NRC considered these?

--Are there any specific problems associated with the use of PRA in the reassessment of the Indian Point plants?

We agreed to divide our review into two phases, with phase one concentrating on PRA techniques as they apply to the Indian Point safety study and phase two addressing the more general aspects of NRC's application of PRA. On February 28 and April 28, 1983, we briefed your staff on problems associated with the use of PRA at the Indian Point plants. This letter summarizes the information we provided at those briefings. Additional information used during the briefings is provided in appendixes I and II. We have initiated phase two of our review and will make any findings available when completed.

The Indian Point PRA is a comprehensive risk assessment which assesses plant systems performance, the ability of the plant to contain radioactivity, and the consequences of potential accidents. While many analysts consider the Indian Point PRA to be the state of the art in risk assessment, it suffers from the same fundamental problems as all PRAs: uncertainty and incomparability of results. Also, although the study identified the dominant contributors to risk, it did not identify the precise level of risk from operating the Indian Point nuclear powerplants.

OBJECTIVE, SCOPE, AND METHODOLOGY

The objective of phase one of our review was to examine the problems of the probabilistic risk assessment of the Indian Point plants. To respond to your request, we focused phase one on the methodologies and limitations of PRA and the Indian Point study as identified and explained by PRA experts and peer reviewers of the study. The results of phase one of our review, as reported herein, represent neither our assessment of the state of the art of PRA nor NRC's use of risk assessment. We will address these issues separately in phase two of our review.

To accomplish our objective, we reviewed the Indian Point PRA; related peer review comments and studies; laws and proposed legislation relating to risk assessment; proposed Department of Energy and NRC guidance on nuclear powerplant safety; numerous scientific articles, papers, and presentations; and previous GAO studies. We interviewed representatives from a variety of organizations that have made significant comments on PRA or on the Indian Point Probabilistic Safety Study. These included the Indian Point owners (Consolidated Edison Company of New York, Inc., and the Power Authority of the State of New York); public interest groups (Union of Concerned Scientists, Friends of the Earth, New York Public Interest Research Group, and the National Audubon Society); the Sandia, Brookhaven, and Oak Ridge National Laboratories; federal agencies (Department of Energy and NRC) and the Advisory Committee on Reactor Safeguards, a statutory body of advisers to NRC.

We did not attempt to independently assess the quality, reliability, or validity of either the Indian Point Probabilistic Safety Study or peer review of the study. We also did not evaluate or assimilate rebuttals to the review criticisms from the study's authors because of the time constraints of our review and the owners' reluctance to provide this information while the Indian Point safety hearings being conducted by NRC's Atomic Safety and Licensing Board were in progress. However, we based our summary of the important peer review findings of the Indian Point study on revised peer review reports that reflect the licensees' views and comments.

As agreed with your office, we did not obtain formal agency comments for phase one of our review. However, to ensure technical accuracy, we obtained unofficial comments from cognizant NRC representatives regarding the peer review of the Indian Point study and its findings. Where appropriate, these comments have been incorporated in the report.

Our review was performed during the period from July 1982 to March 1983 in accordance with generally accepted government audit standards, except that we did not obtain formal agency comments.

WHAT IS PRA?

Probabilistic risk assessment is a method of systematically examining complex technical systems, such as nuclear powerplants, to identify their associated public health, environmental, and economic risks. To assess risk, analysts attempt to quantify probabilities and consequences of accidents as accurately as possible in order to determine mathematical expressions that are realistic. Probabilistic assessment of nuclear powerplants is relatively new; significant use of PRA techniques in this area has occurred only during the past 10 years.

Since 1975, more than a dozen PRAs of varying scope have been performed. Many experts believe that PRA methodology and its application have improved since the first study, leading to more thorough and consistent risk assessments. Several groups that investigated the March 1979 Three Mile Island accident recommended increased use of PRA techniques to enhance safety reviews, and proponents of PRA say that important qualitative insights can be gained in spite of the uncertainties in the quantitative results. Identification of significant contributors to risk and a better understanding of plant design and operation are cited as valuable PRA products.

However, others believe that while PRAs do yield numerical estimates and are thus "quantitative," the estimates are so imprecise and subject to manipulation as to be virtually useless in decisionmaking. Given such controversy, it is not surprising that the role of PRA in the regulatory process has not been defined as yet.

PRAs, by their nature, are statements of uncertainty. They estimate what events and combinations of events are possible and how they might occur. Completeness, sufficiency, and reliability of data, analyst assumptions, and model validity are areas of uncertainty cited by experts. Such uncertainties diminish the precision and reliability of bottom-line risk estimates.

In addition, the results of individual PRAs may be difficult to compare to other studies. Such comparisons may be desired to determine the relative risk among plants or to measure the risk of a plant against a predetermined standard or goal. PRA comparisons are limited by a lack of standardized PRA methodology; variations in scope, data, and assumptions; and the uncertainties of PRA results.

THE INDIAN POINT PRA

The study is a comprehensive risk assessment including plant systems, containment, and consequence analyses. It also ventures into areas where little work had been previously done, such as external events analysis, which includes earthquakes, winds, and fire.

Assessing the degree of risk and identifying the major contributors to risk resulting from the operation of the Indian Point nuclear powerplants were purposes of the safety study. The results of the Indian Point study focused on public health effects rather that on environmental or economic effects. According to the study, the likelihood of damage to public health is remote. For example, the likelihood of an accident which causes any adverse public health consequences is one in 1,000 years of reactor operation. Further, the study predicted with 90 percent confidence that the likelihood of an accident (1) causing any acute fatality is one in 1.7 million years or (2) resulting in 100 or more latent cancer deaths is one in 1,400 years. Several other health effect parameters, such as illness and non-fatal cancers, were also addressed in the study.

In addition to assessing potential public health effects, the Indian Point PRA addressed the likelihood of nuclear fuel melting (core melt) in each reactor during an accident. Although the melting of the nuclear fuel, by itself, may not constitute a threat to public health, it is one element in determining the risk from operating the plants. The study estimated that the mean likelihood of the nuclear fuel melting at Indian Point 2 is one in 2,100 years of reactor operation, with external factors like earthquakes, wind, and fire contributing most to the event. In contrast, the study estimated that the mean likelihood of the nuclear fuel melting at Indian Point 3 is one in 5,300 years of reactor operation with internal factors, like loss of coolant accidents, contributing most to this event. Structural differences account, in part, for the disparity in results between the two plants.

PEER REVIEW OF THE INDIAN POINT PRA

Technical peer review is essential to assure the quality and credibility of a complex study like the Indian Point PRA. The purpose of peer review is to examine and evaluate the appropriateness of a study's data, models, judgments, assumptions, and conclusions. Various individuals and organizations, including NRC, the Advisory Committee on Reactor Safeguards, and public interest groups, such as the Union of Concerned Scientists, have reviewed parts or all of the Indian Point PRA.

Many of the preliminary peer review comments and criticisms of the Indian Point PRA indicated that although the study represents and even extends the state of the art in PRA methodology and knowledge, it also suffers from the same limitations in data, modeling, and completeness as other PRAs.

Peer reviewers found that uncertainties in the study's data, models, and analyses were greater than the study's analysts estimated. While reviewers identified several significant errors, omissions, and critical judgments that may affect risk estimates, they emphasized that the overall uncertainties associated with the study render precise risk estimates unreliable. Thus, the Indian Point study may have either underestimated or overestimated the actual risk of the plants.

The Indian Point PRA can be summarized by noting that although the study analysts were able to identify the dominant contributors to risk at the plants, they were unable to identify the precise level of risk associated with the plants. Thus, it appears that it would be extremely difficult to compare the risk of Indian Point to that of other plants, since PRAs performed on those plants would also be subject to considerable uncertainty in their risk estimates.

As agreed with your office, we plan to send the report to the Nuclear Regulatory Commission and other interested parties on the date it is issued.

Sincerely,

for J. Dexter Peach
Director

Contents

ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
GAO	General Accounting Office
NRC	Nuclear Regulatory Commission
PRA	probabilistic risk assessment

PROBABILISTIC RISK ASSESSMENT

WHAT IS PRA?

Probabilistic risk assessment (PRA) is a method of systematically examining complex technical systems, such as nuclear powerplants, to identify and measure their associated public health, environmental, and economic risks. To assess risk, it is necessary to measure both the likelihood that an accident will occur and the level of damage or loss that will result. These two essential components of risk are referred to in PRA terms as probability and consequences.

PRA methods provide for mathematically quantifying risk based on calculated probabilities of component and human failure and the anticipated consequences of these failures, whether they occur either singly or in combination. PRA addresses three basic questions:

--What could go wrong?

--How likely is it that this will happen?

--If it happens, what are the consequences?

The PRA practitioner attempts to quantify probabilities and consequences as accurately as possible in order to determine mathematical expressions of risk that are realistic. When risks have been quantified in a consistent manner, they can be compared to determine which risks appear to be the greatest and what the major contributors to risk are.

Probabilistic risk assessment of nuclear powerplants is relatively new; significant use of PRA techniques in this area has occurred only during the past 10 years. PRA is often described as an immature and rapidly changing field due to the ongoing development of new techniques and refinement of existing methods.

PRA Scope

The scope of a nuclear powerplant PRA can vary greatly depending on the objectives of the study and the resources available. The three general levels of scope are:

1. Plant systems analysis.

2. Plant systems and containment analysis.

3. Plant systems, containment, and consequence analysis.

These are sometimes referred to as level-one, -two, and -three PRAs. In addition, external events, such as earthquakes and fire, may or may not be included in a PRA.

Plant systems, containment, and consequence analyses

A plant systems analysis, a level-one PRA, is an examination of the design and operation of the powerplant. It identifies how, when, and why accidents could occur within a plant and what the likelihood of such occurrences are.

Containment analysis is an examination of the physical processes of an accident and their effect on the reactor vessel, which is the immediate reactor container, and on the steel and concrete containment building which surrounds the reactor vessel, steam generator, and much of the reactor cooling system. Should an accident or malfunction occur which releases radiation from the reactor vessel or cooling system, the main purpose of the containment building is to prevent the escape of that radiation to the outside environment. The containment analysis predicts how and when containment can fail and what radiation could be released if such failures occurred. Containment analysis is done in addition to plant systems analysis. When combined, these two analyses constitute a level-two PRA.

Consequence analysis predicts the movement of radiation throughout the environment after it has been released (i.e., after containment failure) and estimates the public health and economic effects of the release. A consequence analysis is done in addition to plant systems and containment analyses. Only this third level of PRA permits an overall assessment of plant risk, since it considers both elements of risk, the probability that an accident will occur, and the consequences of such an occurrence.

External events

Each of the above three levels of scope may or may not include an analysis of the effects of external events, such as fires, floods, earthquakes, and storms. Analyses of external events require consideration of factors which may not have otherwise been included in the PRA, such as numerous concurrent failures and the magnitude of an event versus its frequency of occurrence. For example, an earthquake could damage many components simultaneously as well as disrupt plans for evacuating nearby populations. Also, the magnitude of the earthquake must be considered in addition to its frequency of occurrence.

Analyses which include external events tend to be less certain than those that do not because of greater complexity, less experience in this area of analysis, and a lack of historical data. This results in greater reliance on subjective input, such as engineering judgment and expert opinion. However, if a PRA does not consider external events, it is incomplete.

GENERAL PRA METHODOLOGY

Although PRA methodology as applied to nuclear powerplants is a relatively new and evolving area, certain general methods of analysis are widely used and accepted. The first major application of many of these techniques was the Reactor Safety Study (NRC report WASH 1400, also known as the Rasmussen Report) which was completed in 1975. Since then, more than a dozen PRAs of varying scope have been performed. Many experts believe that PRA methodology and its application have improved since that first major study, leading to more thorough and consistent risk assessments, but that further improvement and standardization are needed. Toward this goal, NRC, recognizing "a need for technical guidance on methods and procedures," released a draft of a PRA Procedures Guide in April 1982. This guide, which is a joint government and industry effort, includes information on many aspects of PRA with emphasis on the principal methods of analysis now used. Much of the following summary of PRA methods is based on this guide.

Collection of information

The entire PRA process requires vast amounts of information that, depending on the scope, can include

--plant design and operating information, such as drawings of piping and electrical systems and written operating procedures;

--generic and plant-specific data concerning frequency of initiating events and component reliability, such as NRC-compiled data summaries on pumps, valves, and other plant components; and

--site-specific meteorological, topographical, and population density information.

Sufficient accurate data improves the reliability and precision of probabilistic risk assessments.

Detailed analysis of large amounts of data is possible through the use of computer programs which have been developed for use in various segments of PRAs.

Plant systems analysis

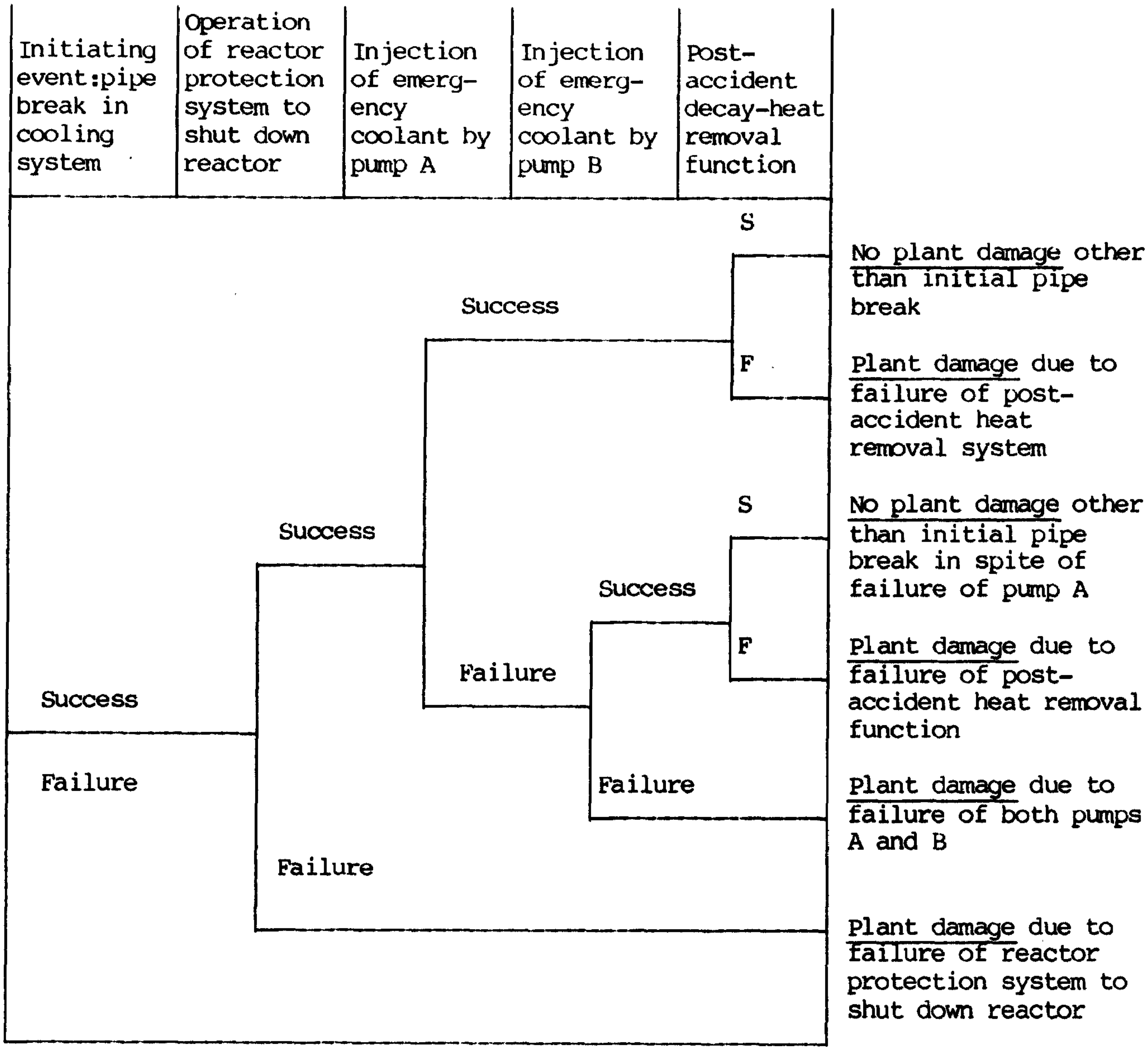
This phase of PRA begins with a systematic search for contributors to risk. Two methods of analysis accomplish this and provide a graphic display of the contributors and their interrelationship. The first is event tree analysis, which identifies the sequences of events that may result in an accident. The second is fault tree analysis, which determines how failures in safety systems may occur.

Event tree analysis

Event tree analysis begins with an attempt to identify all conceivable events that could precipitate an accident, such as a pipe break or loss of power to a necessary plant system. These events are referred to in PRA terminology as "initiating events." Next, all significant sequences of events, or "scenarios," that could follow each initiating event are developed. Each scenario varies depending on the assumed success or failure of mitigating safety systems throughout the sequence. Redundant safety features, such as multiple pumps and physical barriers, are built into nuclear powerplants so that the failure of one component, barrier, or mitigating system alone will not cause an accident. If these backup systems in a particular scenario succeed, then the scenario will be terminated before it culminates in an accident.

An example of a simple event tree adapted from the PRA Procedures Guide follows.

A Simple Event Tree for a Single Initiating Event



In the above example, it is assumed that:

- Either emergency coolant pump A or B is sufficient for successful emergency cooling.
- Failure of the reactor protection system to shut down the reactor will automatically result in plant damage. In this case, it is unnecessary to consider the other three events.
- Failure of both pumps A and B will necessarily result in plant damage.

Fault tree analysis

The construction of fault tree diagrams is a method of system modeling that displays the various ways that a mitigating safety system can fail. Each safety system failure that was identified in the event tree analysis as contributing to an accident is investigated to determine how faults (i.e., failure or malfunction of a component) within that system could contribute to failure of the entire safety system. This analysis should consider component failure, human error, maintenance and testing activity, potential system interaction, and common cause contributors.

Human errors identified in the fault tree diagram are analyzed separately in a human reliability analysis which includes its own event trees and assignment of human-error probabilities. The results of this analysis are then integrated into the fault tree analysis.

After fault trees have been developed, the relationship of various component failures can be roughly determined by grouping the basic, or most elementary, events into minimal failure sets. These minimal failure sets, also referred to as "minimal cutsets," represent the minimal number of fault event combinations that can lead to a given accident sequence. Put another way, all components of at least one minimal cutset must fail in order to cause the failure of a particular safety system; if only one component is restored, the system will succeed. Identification of minimal cutsets facilitates the quantification process and the evaluation of scenarios which could result in the failure of all components of an entire minimal cutset.

Accident sequence quantification

To quantify the likelihood that an accident sequence will occur, e.g., that a scenario will culminate in core damage or core melt, frequencies of occurrence are assigned to initiating events and to failures or human errors identified in fault tree analyses. Frequencies are based mainly on component reliability information gathered from plant operating records, generic information, and expert opinion. Initiating events and success/failure models are combined and quantified with the help of computers to determine frequencies of occurrence of entire accident sequences.

Containment analysis

The second phase of a full-scale PRA is an analysis of the physical processes which may occur following core damage or meltdown and possible escape of radiation from the containment building. Experience in this area is limited, and the state of knowledge concerning the physical processes which could occur following core damage is changing. For these reasons, the accepted methodology in this area is in a "high degree of flux."

The containment analysis considers several stages of events within the containment building that may lead to containment failure. These include:

--Conditions before core melt, such as pressure within the containment building.

--Events related to in-vessel phenomena. This refers to events within the reactor vessel during and after core damage.

--Events related to phenomena after the reactor vessel fails.

--Events related to the disposition and cooling of debris within the containment building. This concerns the behavior and effect of the radioactive materials after release within the containment building, but before release to the outside environment.

Containment event trees

Containment event trees are extensions of the plant systems event trees that were developed in level one of the PRA. However, while the plant systems analysis addressed questions of safety systems' success or failure, the containment event trees ask yes or no questions concerning activity within the containment building, such as "Is water present in the reactor cavity at the time of vessel failure?" It is these final branches leading to containment failure that represent an accident that could result in adverse public health consequences due to a release of radiation.

Since the state of knowledge concerning phenomenology within containment is changing, the quantification of containment event trees involves subjective judgments.

Establishment of release categories

After accident sequences have been identified, analysts determine what amount and type of radiation could be released as a result of each accident and what the mode of the release would be. For example, release could occur as a steam explosion or a slow leak into the atmosphere, or the core could melt into the ground beneath the containment building.

Since the analysts may identify hundreds of accident sequences, it may not be practical to perform release analyses for every sequence individually. For this reason, the sequences may be grouped according to similar characteristics into release categories. This simplifies the analysis by assuming that the radiation release for all sequences within each category will be the same, and it allows accidents to be organized by severity of release. The establishment of release categories is a subjective process, since generic categories have not been developed. Two examples of categories that were used in a recent PRA are

--filtered vented release, in which the release is partially decontaminated as it passes through a filtered vent system, and

--steam explosion with sprays, in which a steam explosion has occurred within the containment building and the water spray system, a safety feature designed to reduce the radiation that would be released, has functioned.

Dominant release categories can be displayed in a table which shows the type of radioactive release and frequency for each category. This information is then provided to the analysts who will perform the next level of the PRA, the consequence analysis.

Consequence analysis

Consequence analysis involves the environmental transport and disposition of released radiation and the resulting effects on humans and the environment. Many variables must be considered:

--Weather conditions, wind direction, and topography of the surrounding terrain affect the dispersion of released radiation. For example, wind may carry it far from the plant or rain may bring it down to Earth.

--The location and density of nearby populations determine the number of people that could be exposed to released radiation. More people would be affected if radiation settled over a highly populated city than over a rural area.

--The quantity and mode of exposure to radiation determine the severity of adverse health effects that are likely to result in a given population. Dosages can be measured and used to estimate specific health effects, such as fatalities, cancer, and genetic effects.

--Mitigating circumstances, such as evacuation of the nearby population or the availability of shelter, will affect the severity of human exposure.

The analysis is done by consequence modeling, an area that, like most other PRA methods, is still under development and requires some subjective input. Several computer programs are available for modeling in this area, but, according to the PRA Procedures Guide (April 1982, Review Draft), in the United States there are four that can perform a complete consequence analysis. They are CRAC, used in the Reactor Safety Study, and three offshoots, CRAC 2, CRACIT, and NUCRAC, which are refinements and expansions of CRAC.

Presentation of results

The results of a full-scope PRA, sometimes referred to as a "level-three" PRA, integrate the results of the plant systems analysis, the containment analysis, and the consequence analysis. Results can be presented in tables listing major scenarios and identifying their release categories, contribution to core melt, likelihood of causing damage to the public health, and other information of interest. Some information can also be displayed in graphic form.

In addition, uncertainties and their effects on the risk results must be considered and in some way presented with the results. PRA involves uncertainty at all levels. For example:

--Data may be unavailable, incomplete, inappropriate, or biased.

--Modeling may not be a good representation of reality, or it may be improperly used.

--The analysis may be incomplete. The analyst cannot evaluate all contributors to risk perfectly and exhaustively, so there is always a possibility that an important event or factor has been overlooked.

Estimates of uncertainty are made at many levels of the PRA, often by statistical analysis resulting in the establishment of a range rather than a specific number of a frequency or damage level. These assessments of uncertainty can be combined by use of mathematical formulas to determine an overall estimate of the uncertainty in the final risk results. However, the field of uncertainty analysis as part of PRA is changing. As with other PRA methods, experience is limited, and there is no "generally accepted rigorous mathematical basis" for analyzing uncertainty.

The most widely used quantitative measure of uncertainty has been the idea of "confidence bounds" or "confidence levels." The confidence levels express the analysts' degree of confidence that the risk estimates are realistic based on such things as the quantity and reliability of data and the quality of computer codes used in the PRA. Often three confidence levels are displayed representing upper and lower bounds and a "best" estimate falling between the two. The range between the upper and lower bounds is referred to as the "uncertainty band."

Examples of partial results from a recent PRA presented in tabular and graphic form are shown and explained on the following pages.

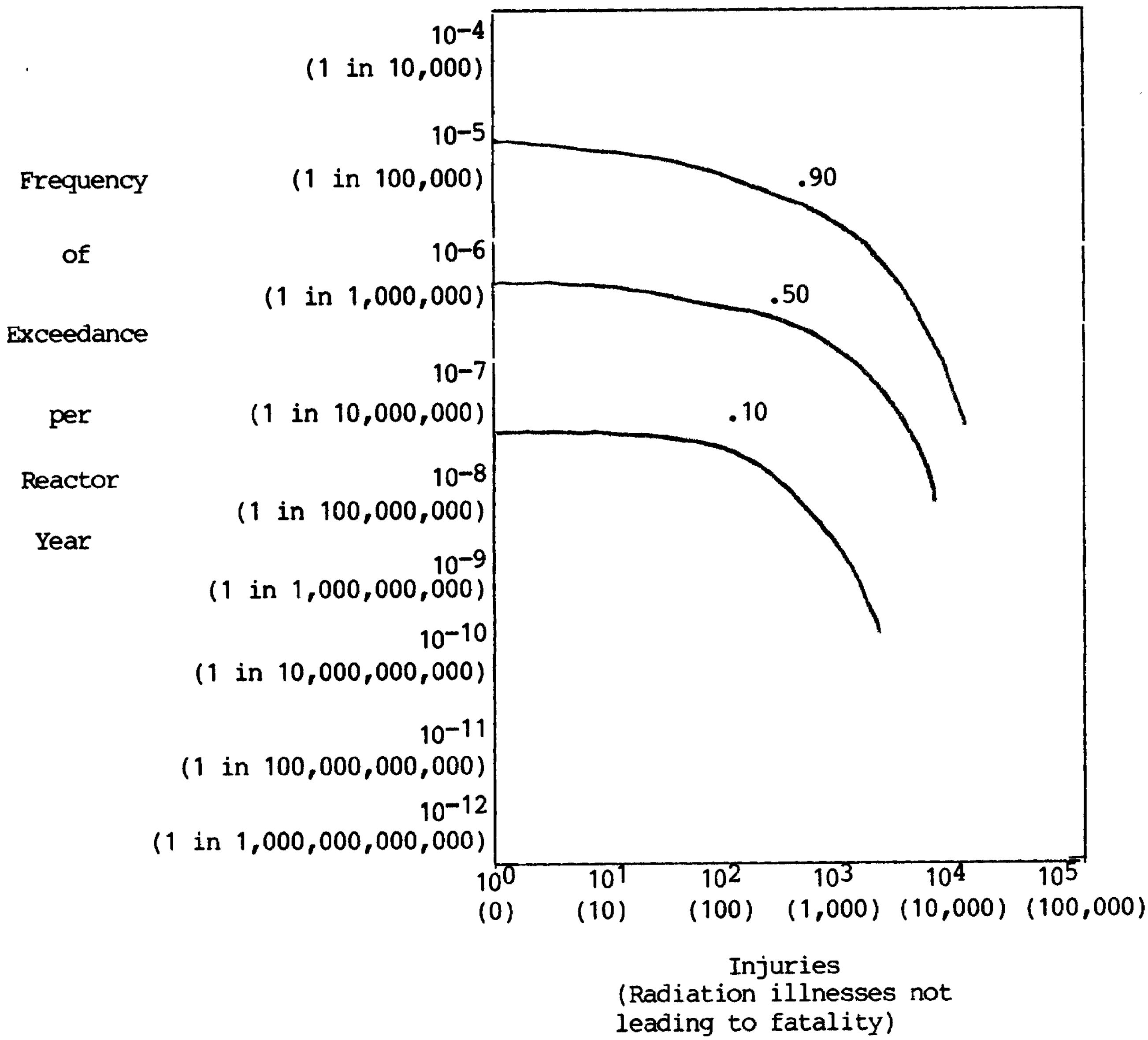
The following table is an example of one way that certain PRA results can be presented. This particular table is part of a larger table that was included in a recent PRA. The wording of some captions has been changed to make it more understandable. Also, frequencies, which are expressed in scientific notation, have been explained in parentheses (rounded).

COMPARISON OF CORE MELT AND RELEASE FREQUENCY CONTRIBUTIONS OF MAJOR SCENARIOS

Sequence	Mean annual frequency (contribution to core melt)	Rank with respect to core melt	Mean annual frequency of a release which would cause early fatalities	Relative rank with respect to early fatalities release frequency
Seismic: Loss of control or power	1.4 X 10^{-4} (1 chance in 7000 reactor years)	1	2.8 X 10^{-8} (1 chance in 35,700,000 reactor years)	3
Fire: Specific fires in electrical tunnel and switchgear room causing RCP seal LOCA and failure of power cables to the safety injection pumps, containment spray pumps, and fan coolers	1.4 X 10^{-4} (1 chance in 7000 reactor years)	2	2.8 X 10^{-8} (1 chance in 35,700,000 reactor years)	4
Fire: Specific fires in electrical tunnel causing RCP seal LOCA and failure of power cables to all MCCs, safety injection pumps, RHR pumps, and containment spray pumps	5.0 X 10^{-5} (1 chance in 20,000 reactor years)	3	5.5 X 10^{-9} (1 chance in 181,800,000 reactor years)	5
Turbine trip due to loss of offsite power: Failure of two diesel generators, RCP seal LOCA, and failure to recover external AC power until after 1 hour	3.0 X 10^{-5} (1 chance in 33,000 reactor years)	4	3.0 X 10^{-9} (1 chance in 333,300,000 reactor years)	8

APPENDIX I

The graph below shows "risk curves" representing the frequencies of many accidents and the damage levels (in this case, the number of injuries) which would result from their occurrence. There are three risk curves, referred to as a family of curves, representing the 10 percent, 50 percent, and 90 percent confidence levels. This graph shows that the number of injuries is highest for the scenarios that are least likely to occur.



LIMITATIONS OF PRA

Although probabilistic risk assessment has been praised as a good method for the systematic examination of nuclear powerplant risk and as a basis for risk management, its use is limited because PRA results are uncertain and difficult to compare. Uncertainties diminish the precision and reliability of bottom-line risk estimates, while incomparability hinders determinations of relative risk.

Further, PRAs are wide-ranging, complicated studies that are difficult to understand and evaluate. For this reason, results may be misinterpreted, leading to mistaken perceptions of which plants are the safest.

Uncertainties in PRA

By their nature, PRAs are statements of uncertainty. They identify what events and combinations of events are possible and how often they may occur. They do not predict when an event may occur. For example, if we could be certain that a particular scenario would result in an accident and if we could be certain that this scenario would occur once in every 50 years, we still would not know in which of the 50 years the scenario would occur. PRA only provides us with the probability of an accident's occurrence.

If we could eliminate all uncertainty, we could eliminate all risk. For example, if we were certain that a particular scenario would result in an accident and if we were certain that this scenario would occur on January 29, 1999, we would shut down the plant before that date or take steps to eliminate the possibility of that scenario's occurrence. Unfortunately, such complete certainty in future events is impossible.

Almost all aspects of PRA contain some element of uncertainty, but the level of uncertainty varies. A PRA analyst may be very certain that a particular valve will function properly 98 percent of the time because plant records show that it has functioned properly 98 percent of the time for the past 5 years. However, reliability of another valve may be less certain because there are no plant records or because the valve is rarely called on to function. In the case of the second valve, the analyst may supplement whatever plant-specific data is available with generic valve reliability information and engineering judgment.

Uncertainties that affect the precision and reliability of PRA results may reduce the usefulness of PRA for decisionmaking purposes. Experts cite the following specific areas of uncertainty in PRA.

--Completeness.

--Sufficiency and reliability of data.

--Assumptions made by study analysts.

--Validity of models used.

Completeness

It is difficult, if not impossible, to ensure that a PRA is complete, i.e., to ensure that all events and combinations of events which could lead to an accident have been considered. There is always a possibility that a scenario has been overlooked or unimagined. In addition to unintentional omissions, certain events may be purposely omitted.

Unintentional omissions can include odd or unknown events, or can result from the complicated nature of PRA. Hundreds of thousands of scenarios may be considered in one study, and the chance that a significant combination of events may have been overlooked cannot be completely eliminated. This is especially true in the analysis of common cause events which trigger multiple failures simultaneously or sequentially. In addition, due to the large number of possible failures that may be considered in a PRA, low-level failures which appear insignificant may be eliminated to simplify the analysis. It is possible that significant low-level failures could be inadvertently eliminated in this process.

On the other hand, some events are purposely omitted for various reasons. Examples include the following:

--External events may be omitted because they introduce substantial additional uncertainty into the PRA results.

--War may be omitted on the premise that its effects would overshadow any powerplant damage.

--Sabotage may be omitted because there is no basis on which to measure this risk, or because analysts assume that its worst consequences could not exceed the worst consequences of accidents.

Sufficiency and reliability of data

Insufficient reliable data contributes to the uncertainty of PRA. Appropriate data may be scarce due to a lack of experience, as is the case with unusual events and failures, or to a lack of understanding, as is the case concerning phenomena within the containment building during and after core melt. In such situations, analysts must use their own judgment in deciding what data to use and what statistical methods to apply.

Some events and failures which affect nuclear powerplant risk are infrequent or have never occurred. There is little accumulated data on the reliability of components which are rarely called on to function. Some potentially disastrous events, such as severe earthquakes near nuclear powerplant sites, have been rare in recent history, so that no historical data is available on the frequency and effects of such occurrences at nuclear powerplants. In these situations, analysts must rely on generic data, simulations or mathematical models based on scientific theory, and small-scale experiments. Data from such sources is less certain to represent reality than plant-specific data based on operating experience.

There is a particular shortage of data on human reliability, especially on how operators react under stress. As a result, PRA analysts take a conservative approach by usually assuming that humans either do nothing to help or make errors which compound the situation. Creative operator intervention and mitigating efforts not described by plant emergency procedures are usually not considered.

Another source of data uncertainty is that different statistical interpretations of the same data may result in different conclusions. Analysts choose methods according to factors such as the availability of data sources and the desired treatment of uncertainties. There is no single set of statistical methods which must be used in PRA.

Assumptions made by study analysts

The possibility that analysts will make incorrect assumptions contributes to PRA uncertainty. Assumptions may simplify a study or limit its scope, or assumptions may be necessary in areas that are not well understood. Subsequently, such assumptions may be questioned by other PRA experts or disputed by new evidence. Existing PRAs have included assumptions

--concerning phenomenology within the containment building, which were subsequently criticized by other PRA experts;

--that an analysis of internal flooding at one plant would also apply to a similar plant, making a repetition of that analysis unnecessary; and

--that the plant was designed and constructed properly, so that design and construction errors need not be considered.

In areas that are not well understood or where little data exists, assumptions may be necessary before analysts can proceed with the study. Such areas include human behavior, external events, and phenomenology within the containment building during and after core melt. For example, in the fire analysis segment of a recent PRA, analysts assumed that:

--Plant damage would result only from actual burning. Damage due to hot gases was not considered.

--Operator actions would not be hampered by confusion due to a fire.

These assumptions were subsequently questioned by reviewers of that PRA, who cited recent test results as evidence that hot gases resulting from fire could cause additional damage and said that the assumption concerning operator actions was not conservative.

<u>Validity of models used</u>

The reliability of a PRA is affected by how well models used in the study represent reality; models which appear to be accurate representations of reality increase the analysts' certainty in PRA results. Models are used in all segments of PRA to display the relationships among system components and events and to facilitate the quantification process. They may describe plant systems (e.g., event trees and fault trees); phenomena within the containment building; or consequences, such as transport of radioactivity throughout the environment or the geography and populations surrounding the plant. Analysts may develop models for a specific PRA or modify available generic models.

Modeling problems that contribute to the uncertainty of PRA include the following:

--Computer models, which have been developed for many aspects of PRA, are sometimes simplified representations of reality or have not been in use long enough to be validated.

--Criteria for determining the success or failure of plant systems may be inaccurate. Partial failures are difficult to model because binary coding used in the computerized quantification process provides only two alternatives, i.e., success/failure or yes/no. Subtle distinctions cannot easily be made.

--Phenomenology within the containment is not well understood and, therefore, is difficult to model.

--Common cause failures, which include external events, involve many concurrent failures which are difficult to model.

Limited comparability of PRAs

In addition to the uncertainties in individual PRAs, the results from studies of two or more plants may be difficult to compare. Such comparison may be desired to determine the relative risk among plants or to measure the risk of a plant against a predetermined standard or goal. PRA comparisons are limited by

--a lack of standardized PRA methodology;

--variations in scope, data, and assumptions; and

--the uncertainties of PRA results.

Lack of a standardized PRA methodology

The lack of standardized PRA methodology makes it difficult to evaluate the quality and objectivity of a PRA and to compare one study with another. No single, widely accepted methodology exists for doing a PRA of a nuclear powerplant. Experts disagree on which methods are best and, often, there is no way to prove who, if anyone, is right or wrong. PRA practitioners must use their own judgment to choose the methods and data which are best suited for each study. As a result, two equally competent teams of practitioners could do a PRA of the same plant, and the results could differ substantially.

In some areas, such as the analysis of external events, there are few precedents, so that analysts may be compelled to modify existing methods to suit their study. The accuracy of such methods in predicting these rare events has not been and probably cannot be proven.

In addition to the discrepancies of PRA results which may arise due to legitimate differences in opinion among analysts, the lack of a standardized methodology can provide opportunities for analysts to choose methods which slant results toward a predetermined assessment. For this reason, the objectivity of a PRA, or at least the perception of objectivity, is diminished.

The need for some standardization has been recognized by NRC and others, but NRC's Advisory Committee on Reactor Safeguards has said that it is still too early in the development of PRA to provide "prescriptive rules." PRA experts fear that such endorsements may "lock in" procedures that are not always appropriate and discourage improvements and innovations.

However, some steps have been taken toward standardizing PRA methodology. In April 1982, a review draft of the PRA Procedures Guide was released. Prepared under the auspices of the American Nuclear Society and the Institute of Electrical and Electronic Engineers with a grant from NRC, this document outlines acceptable PRA methods but still leaves many choices among alternatives to the PRA analysts. In addition to this effort, NRC has developed another procedures guide tailored for use in the National Reliability Evaluation Program, a limited PRA program that NRC plans to integrate with other NRC evaluations of operating reactors.

Variations of scope, data, and assumptions

Since the scope, availability of data, and assumptions made by analysts affect the results of PRAs, variations in these elements reduce the comparability of PRA results.

A major variation in the scope of existing PRAs is that some have included considerations of external events and others have not. Such a variation can significantly affect results, as indicated by a recent PRA which identified external events as major contributors to risk.

Variations in the availability of appropriate plant-specific and generic data may affect the analysts' confidence in a study's results. These variations, both in sources of data and in confidence levels, increase the disparity among PRA results.

Differences in assumptions made by analysts, as discussed in the previous section on uncertainty, also add an element of inconsistency to PRA results.

Comparison is hindered by uncertainty of PRA results

Comparison of PRA results may be of little value due to their uncertainty. In addition, such comparisons may be misleading when single point estimates are used, since the effect of uncertainties in the results is not apparent. The following example displays the significance of uncertainty in such comparisons.

Recent NRC staff comments concerning plant safety included a table which allows comparison of various results from existing PRAs. The table shows that two plants have the same core melt frequency of 4×10^{-4}, which can also be stated as once in every 2,500 years. However, text which accompanies the table states

> "* * * the numbers in the table have large uncertainty bounds associated with them. In general, these uncertainty bounds should extend on the order of plus of minus a factor of ten about the values presented."

If uncertainty bounds of plus and minus a factor of 10 are applied to the above core melt frequency, the result is a range between once in every 250 years (4×10^{-3}) and once in every 25,000 years (4×10^{-5}). This means that although the two plants have the same single point estimates for frequency of core melt, these estimates are uncertain, and the actual frequency could fall anywhere within the range between the uncertainty bounds. If the actual core melt frequencies for the two plants fall at opposite ends of the range, the frequency at one plant could be once in every 250 years, while the frequency at the other plant could be once in every 25,000 years, or 100 times less frequent.

Potential misinterpretation of results

PRA results must be interpreted carefully by those who are involved in risk management since studies may be misinterpreted or misused by those who do not understand PRA and its limitations. Because they are complicated, highly technical studies, extensive review by experts is necessary to assess the reliability of a PRA. The validity of some aspects may never be proven, since thousands of years of plant operating experience would be needed to do this. In addition, some experts warn that catastrophic, but extremely unlikely, accidents may receive a disproportionate amount of attention.

Intermediate PRA results, such as core melt frequencies, may not be an accurate measure of plant risk. However, attention is sometimes focused on these results, in part because of NRC's emphasis on accident prevention (i.e., prevention of core melt) and because the plant systems, which must be analyzed to determine core melt frequencies, are better understood than containment and consequence analyses. Core melt frequencies may not indicate public risk because release frequencies and potential damage levels have not been considered.

For example, although PRA results may show that two plants each have, relatively, the same core melt frequency, when release frequencies and consequences are considered, the risk could be greater at one plant than at the other. The chance that a release of radiation will result from a core melt may vary between the plants because of differences in the plant containment system. In addition, the damage levels that could result from a release of radiation could also vary depending on factors such as weather patterns, the size of nearby populations, and the ability of the populations to evacuate or take shelter in an emergency.

Another concern of PRA experts is that rare catastrophic accidents with very uncertain probabilities and consequences will receive a disproportionate share of attention at the expense of less severe, but more likely, accidents. Events such as earthquakes and hurricanes may appear to be major contributors to risk because of the devastating damage that could result from their occurrence at a nuclear powerplant. However, because such events are unusual, they are often not well understood, and there is little or no operational data concerning their effects. For these reasons, analyses of such events may be even less certain than other PRA segments.

THE INDIAN POINT PROBABILISTIC RISK ASSESSMENT

BACKGROUND, SCOPE, AND METHODOLOGY OF THE INDIAN POINT STUDY

The Indian Point Probabilistic Safety Study was an effort "to provide a thorough assessment of public risk" resulting from the operation of the Indian Point units 2 and 3 and to identify the major contributors to this risk. The plant owners, Power Authority of the State of New York and Consolidated Edison Company of New York, Inc., initiated the study mainly in response to questions raised by NRC and special interest groups concerning such risk.

The study was prepared under the supervision of the utilities by Pickard, Lowe, and Garrick, Inc., and Fauske and Associates, firms with experience in PRA and nuclear powerplant safety issues, and Westinghouse Electric Corporation, which supplied the plants' nuclear steam systems and turbines. The effort began in January 1980 and took about 2 years to complete. It involved a team of more than 50 experts, including nuclear engineers, systems analysts, mathematicians, computer specialists, and nuclear plant designers and operators. The final report, consisting of approximately 6,000 pages in 12 volumes, was submitted to NRC in March 1982.

The study has played a major role in hearings currently being conducted by the Atomic Safety and Licensing Board, which was empanelled by the NRC in 1981 to adjudicate certain issues concerning risk associated with the Indian Point plants. The Commission's primary concern is the proximity of the Indian Point plants to highly populated areas and the effect of this proximity on the overall risk of these plants compared to other nuclear powerplants.

Study scope and overview

The Indian Point Probabilistic Safety Study is a comprehensive level-three PRA in that it includes analyses of plant systems, containment, and accident consequences. In this respect, it is a true risk analysis, since it considers both the probability of accidents occurring and the consequences of those accidents. The Indian Point study concentrated on the public health consequences of accidents.

While the analysts who performed the Indian Point study generally followed the methodology which has evolved from the Reactor Safety Study of 1975, they also ventured into areas where little work has previously been done, such as external events analysis. Specific external events analyzed were

--earthquakes,

--fire,

--floods,

--high winds,

--aircraft and other transportation accidents, and

--turbine missiles (broken turbine blades which could cause plant damage).

The following summary of the methodology used in the study is based mainly on the description provided in the study itself. We have made no attempt to dispute, criticize, or validate the analysts' assertions concerning what methods were used. The purpose of this section is to provide a brief and easily understood version of this methodology as outlined in the study.

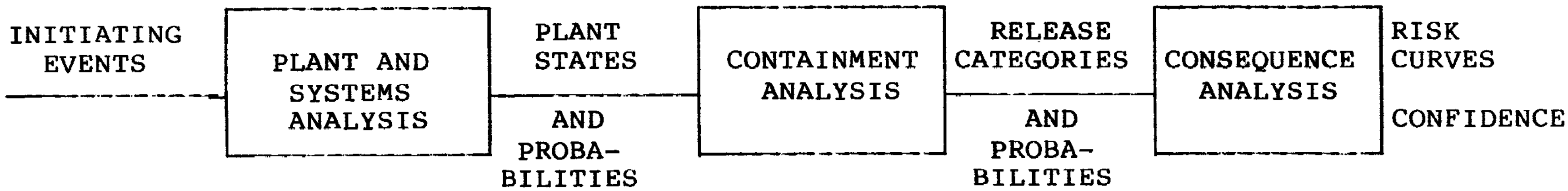
Indian Point study methodology

The Indian Point study consists of three main segments: plant systems analysis, containment analysis, and consequence analysis. For each segment, the analysts developed models which were used to quantify accident sequence frequencies and resulting damage levels. The plant and containment models were developed mainly by event tree/fault tree analysis. A computerized site model was used for the consequence analysis.

The three segments were performed sequentially, and results were categorized at the end of each segment for simplicity and clarity. Accident sequences identified in the plant systems analysis were grouped into plant damage states which served as the initial input for the containment analysis. Possible releases of radiation identified in the containment analysis were grouped into release categories which served as input for the consequence analysis. The final results of the consequence analysis were grouped into health effects categories.

The following diagram shows the relationship of the three segments of the analysis.

PROBABILISTIC RISK ASSESSMENT FLOW CHART



- o SYSTEMS
- o COMPONENTS
- o OPERATIONS
- o TESTS
- o MAINTENANCE
- o HUMAN ERROR
- o EXTERNAL EVENTS

- o CORE
- o REACTOR VESSEL
- o CONTAINMENT
- o CONTAINMENT SYSTEMS

- o METEOROLOGY
- o POPULATION
- o EVACUATION
- o HEALTH EFFECTS

APPENDIX II

Conservatism

The study's analysts state that they took a conservative approach to risk assessment throughout all segments of the analysis by overstating frequencies, damage, and uncertainties whenever the level of these elements was in doubt. For example:

--If an event tree branch point consisted of only two alternatives, one for system failure and one for system success, a partial failure was considered to be a total failure in calculations of accident sequence frequency.

--The analysts modified the confidence bounds suggested in the NRC Human Reliability Handbook to reflect greater uncertainty in data used from that source because of the highly judgmental nature of such information.

--In an attempt to ensure completeness, the analysts included an "other" category in calculations of accident sequence frequencies. This "other" category provides for unknown or unimagined events or conditions that have not been included in the analysis.

Plant systems analysis

The analysts identified possible accident sequences and determined their frequency of occurrence in this segment of the analysis. Hundreds of thousands of scenarios were developed by event tree/fault tree analysis, and those which could result in core melt were identified. The analysts determined the frequency of occurrence of accident sequences based on generic and plant-specific data, general engineering knowledge, expert opinion, and human reliability information.

A statistical method known as Bayes' theorem was used extensively to integrate this data. Bayes' theorem allows analysts to incorporate generic data and expert opinion in frequency calculations. This is useful when there is a scarcity of plant-specific data based on operational experience, as is often the case concerning infrequent events at nuclear powerplants. In such situations, classical statistical analysis may result in imprecise estimates. The use of Bayesian methods may improve the apparent precision of risk estimates.

The analysts grouped the large number of accident sequences identified in this analysis into 21 plant damage states. These categories describe the conditions within the reactor vessel, the way that the core melts, and whether or not the fan coolers and containment spray systems are working. Plant damage states and the frequency of occurrence for each were then provided as initial input for the containment analysis.

Containment analysis

The containment analysis is essentially a continuation of the accident sequences that were identified in the plant systems analysis, since the plant damage states which were established in that analysis were used as the starting points for the containment event trees.

The Indian Point study analysts used 12 containment event trees which graphically display the paths, or sequences of events, leading to a release of radiation. The end points of The containment event trees, which number more that 12,000 for all 12 trees combined, were grouped into 13 radiation release categories. The release categories used were similar to those established in the Reactor Safety Study.

Probabilities were assigned to the branch points on each containment event tree with the help of computer modeling codes and information gathered from consultants, various reports, analyses, experiments, and scientific literature. Analysts then used these probabilities to determine the probability of occurrence for each release category. This information then became the starting point for the consequence analysis.

Consequence analysis

The objective of the consequence analysis was "to estimate the potential health effects associated with each release category." This involved, first, an analysis of the distribution of radiation throughout the environment and, second, an assessment of doses and damage.

The principal analytical tool used for most of the consequence analysis was a series of computer programs originally developed for use in the Reactor Safety Study but extensively modified to incorporate site-specific conditions at Indian Point. The modified series, called the CRACIT code, calculates the dispersion of radiation throughout the environment over measured time periods based mainly on radiation release data and meteorological information. It also calculates potential health effects based on population distribution and evacuation information.

The effects of varying doses of radiation on the public health were calculated for 6 of the 13 release categories. The other categories were not analyzed in detail because preliminary calculations indicated that their frequencies and consequence level were low.

The specific health effects estimated in the Indian Point study were

--fatalities that occur within 7 years after exposure;

--injuries that occur within 7 years after exposure, but do not result in death;

--thyroid cancer cases, which were estimated separately because of their low fatality rate;

--cancer fatalities other than thyroid cancer, usually occurring from 2 to 30 years after exposure; and

--whole body man-rem, a radiation dose measurement, which can be used to estimate genetic effects as well as other health consequences.

These health effects were calculated using data similar to that used in the Reactor Safety Study.

The analysts' uncertainty in the final consequence results was ultimately based on their own judgment after performing sensitivity studies which explored the results of variations in accident conditions. Results are presented in a variety of combinations both in tables and, graphically, as risk curves.

SAFETY STUDY RESULTS

Assessing the degree of risk and identifying the major contributors to risk resulting from the operation of the Indian Point nuclear powerplants were purposes of the safety study. To that end, the study attempted to answer the following three questions which are characteristic of full-scope PRAs:

--What is the likelihood of core melt?

--What is the likelihood of release of radioactivity outside the plant?

--What is the likelihood of damage to public health?

According to the licensees, the answers to those questions constitute the quantitative expression of risk from the Indian Point powerplants.

What is the likelihood of core melt?

The central part of a nuclear reactor that contains the fuel and produces heat is the core. The melting of fuel in a reactor is a core melt. For purposes of the safety study, core damage was considered the same as core melt. If, for example, some portion of the fuel should melt, the molten fuel could melt through the reactor vessel and release large quantities of radioactive materials into the containment building. However, a core melt by itself does not necessarily constitute a threat to public health because the plant's containment building is designed to prevent significant radioactive releases into the environment.

The authors quantified the likelihood of core melt by drawing probability curves which expressed their knowledge about the melt frequency. From these curves they extracted median and mean core melt frequency estimates. The following table shows the median and mean[1] core melt frequency estimates for the Indian Point (IP) plants.

Frequency of Core Melt
(occurrences per reactor year)

Measure	IP-2	IP-3
Median	1 in 2,500	1 in 11,000
Mean	1 in 2,100	1 in 5,300

Because the Indian Point plants are separate facilities, structural differences account, in part, for differences in core melt probability. For example, IP-2's containment building has soil backfill on one of its sides. As a result of the backfill, IP-2 containment is more vulnerable to seismic events than IP-3 which has no such backfill. Also, the probability of a seismically induced failure of IP-2's control room is greater than for IP-3 because of structural differences.

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[1]Median and mean are frequently used mathematical estimates in the Indian Point study. The median is the point where an equal number of values are above and below that point. The mean is an average of estimates. Another frequently used estimate is the "best estimate." Best estimates are single value estimates which, according to the study's authors, lead to conservative results.

Major contributors to core melt

Initiating events, as previously discussed, are the beginning points in accident sequences. Initiating events can be divided into internal and external plant events. Internal initiating events, like pipe breaks and valve failures, occur due to events within a plant's systems, components, and interconnected electrical systems. External events, like earthquakes, floods, tornadoes, and fires, occur outside the plant's actual systems.

Certain initiating events are major contributors to core melt at Indian Point. The external events--seismic, wind, and fire--contribute to about 82 percent of the total core melt for IP-2. Of this total, fires account for 43 percent while seismic events and wind represent 30 and 9 percent, respectively. With respect to internal events, loss of coolant accidents (LOCAs) combine to contribute about 10 percent to core melt.

In contrast to IP-2, internal initiating events contribute most to core melt at IP-3. For example, LOCAs combine to account for 61 percent of total core melt, with small LOCAs the major contributor at 45 percent. Fires contribute the most of all external events with 33 percent of core melt for IP-3.

The following table compares the degree to which major internal and external events could contribute to core melt at the Indian Point plants.

Major Initiating Events Contributing to Core Melt at Indian Point

IP-2 Internal events	Percentage of total for plant	IP-3 Internal events	Percentage of total for plant
LOCAs	10	LOCA's	61
External events		External events	
Seismic	30		
Wind	9		
Fire	43	Fire	33
All other internal and external events	8	All other internal and external events	6
	100		100

What is the likelihood of release of radioactivity outside of the plant?

If containment serves its function, a core melt, regardless of cause, would only lead to a small radiation leak with little or no consequence to public health. If containment fails, the amount and type of risk to the public depend on factors such as the amount and type of radiation released, wind speed and direction, how quickly the radioactive material is dispersed, and the effectiveness of protective steps like sheltering and evacuation. Thus, the second question concerns the likelihood that the containment will fail to perform its function in the event of a core melt.

The safety study identified 13 radioactive release categories corresponding to different kinds of containment performance. Six of these categories are important by virtue of either their impact or frequency of occurrence. However, only three releases occur early or rapidly and release enough radioactivity to cause early deaths due, in part, to the inability to notify the public or take protective actions. Yet, these three most serious releases are about 1,000 times less likely to occur than the less serious ones since they require, in addition to core melt, the early failure or bypass of containment. A description of the releases follows:

--In one of the three most serious release categories, containment fails as a result of an earthquake.

--Another of the most serious releases represents containment failure one to three hours after a core melt.

--The third of the most serious release categories represents the bypassing of containment through a rupture in a heat removal system or an early overpressure problem.

--Two of the less serious releases apply to all damaged or core melt accidents in which containment remains intact.

--In the third less serious category, containment fails after a core melt due to a gradual pressure buildup.

Major contributors to release categories

As with core melt, certain initiating events can be major contributors to release categories. At IP-2, external events are the major contributors to two of the three important release categories. For example, earthquakes contribute to at least 76 percent of these two most important release categories. On the other hand, the interfacing system LOCA[2] internal event causes 85 percent of the third release.

At IP-3, one serious release category results solely from the earthquake external event. Internal and external initiators contribute about equally to a second category. Finally, the third most serious release is primarily caused by the interfacing system LOCA internal initiator.

What is the likelihood of damage to public health?

The likelihood that an accident would cause any public health consequences is remote according to the safety study. For example, the likelihood of an accident which causes any adverse public health consequences is one in 1,000 years.

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[2]An interfacing system LOCA occurs after a valve separating a heat removal system from the reactor vessel coolant system fails. This releases coolant and fission products outside containment.

The study uses the following damage indices to quantify the risk to public health in the event of containment bypass or failure:

--Acute or early fatalities. Deaths which occur within a short period of time after exposure.

--Injuries. Non-fatal radiation illnesses.

--Thyroid cancers. Treatable and usually non-fatal cancers which occur over a 30-year period.

--Latent cancer fatalities. Latent deaths from cancers other than thyroid cancers, occurring over a 30-year period.

The safety study highlights the damage indices for three frequencies of occurrence--any public health effect, 100 effects, and 1,000 effects. The results are also presented as "best" and "upper bound" estimates which represent the 50 and 90 percent confidence levels. Upper bound estimates mean that the safety study's authors are 90 percent confident that the frequency of occurrence will not exceed a certain value. For example, at IP-2, the best estimate for any acute death is one in 17 million reactor years of operation. Upper bound estimates indicate that an accident resulting in 100 or more latent cancer deaths is likely only once in 1,400 years of reactor operations.

The following table summarizes the public health effects attributable to the Indian Point plants.

Public Health Effects of Operating the Indian Point Plants

Damage index	Best estimate any effects	Upper bound	Best estimate 100 effects	Upper bound	Best Estimate 1,000 Effects	Upper Bound
(in reactor years of operation)						
IP-2						
Acute fatalities	1 in 17 million	1 in 1.7 million	1 in 100 million	1 in 4.8 million	1 in 1 billion	1 in 29 million
Radiation injuries	1 in 370,000	1 in 59,000	1 in 2.9 million	1 in 290,000	1 in 110 million	1 in 2.9 million
Thyroid cancers	1 in 2,500	1 in 1,000	1 in 5,900	1 in 1,400	1 in 12,000	1 in 2,700
Latent cancer fatalities	1 in 3,000	1 in 1,000	1 in 5,000	1 in 1,400	1 in 10,000	1 in 2,400
IP-3						
Acute fatalities	1 in 83 million	1 in 10 million	1 in 310 million	1 in 45 million	1 in 6.3 billion	1 in 290 million
Radiation injuries	1 in 2.6 million	1 in 330,000	1 in 20 million	1 in 2.4 million	1 in 310 million	1 in 28 million
Thyroid cancers	1 in 12,000	1 in 3,100	1 in 63,000	1 in 7,700	1 in 100,000	1 in 12,000
Latent cancer fatalities	1 in 20,000	1 in 5,000	1 in 55,000	1 in 8,000	1 in 100,000	1 in 12,000

APPENDIX II

PEER REVIEW OF THE INDIAN POINT PROBABILISTIC SAFETY STUDY

Technical peer review is essential for assuring the quality and credibility of a complex scientific study like the Indian Point study because, according to the Risk Assessment Review Group which reviewed the Reactor Safety Study, "no technical person or group is infallible." The purpose of peer review is to examine and evaluate the appropriateness of a study's data, models, judgments, assumptions, and conclusions. While peer review cannot guarantee that a study is absolutely correct, it can determine whether a study meets the generally accepted standards current in its field. The House Committee on Science and Technology recognized the importance of peer review when, reporting out a bill on the use of risk analysis by regulatory agencies (H.R. 6159) in June 1982, it stated that,

> "With respect to the question of quantitative applications of risk analysis, it is evident * * * that caution should be exercised continuously and well into the future, and that strenuous scientific peer review should be applied to a risk analysis before the results or conclusions of that analysis becomes [sic] a key factor in a quantitative regulatory decision."

Although several peer reviewers have reviewed the Indian Point study, the peer review of the study is incomplete and limited.

Reviewers of the study

Various individuals and organizations have reviewed or are reviewing parts or all of the Indian Point study. The major lead groups reviewing the study include the

--Nuclear Regulatory Commission (NRC),

--Advisory Committee on Reactor Safeguards (ACRS), and

--Intervenors to the Indian Point hearings.

The reviewers' purposes and extent of their reviews are discussed below.

Nuclear Regulatory Commission

The NRC staff conducted the most extensive review of the Indian Point study as part of its standard review of PRAs in general. NRC also reviewed the study and presented testimony on the risks of the Indian Point plants before the Atomic Safety and Licensing Board hearings on the safety of Indian Point during this past winter.

To prepare its testimony on the risks of the plants, NRC reviewed and analyzed the three basic components of the Indian Point PRA: accident sequences, containment, and consequence analyses. NRC contracted with the Sandia National Laboratories to review the study's accident sequences and revise the study's plant damage state frequencies as necessary. The NRC staff, with support from the Brookhaven National Laboratory, then calculated their own risk estimates of the Indian Point plants using the revised plant damage state frequencies and containment and consequence models independent of the study.

The Sandia National Laboratories reviewed the principal component of the Indian Point study--the plant systems analysis. Sandia conducted what it called a "limited review" of the systems analysis and external event analysis of the study and published a draft report of its review in August 1982 and a final report in December 1982 entitled "Review and Evaluaton of the Indian Point Probabilistic Safety Study" (NUREG/CR-2934). The final report was the result of 6 months of review by Sandia personnel with contractor support and represents the single largest review of the study completed to date.

Sandia analysts reviewed the plant systems analysis, external event analysis, and accident sequences of the study, focusing on the analyses and sequences which the study determined to be most important or dominant. The Sandia analysts also reviewed the "basic building blocks" of the study--initiating events, fault and event trees, human errors, data, common cause events, and accident sequence analyses--searching for significant omissions, errors, uncertainties, and critical judgments.

The NRC staff and Sandia reviewers assumed that the study accurately represented the Indian Point plants as designed and built. For example, the reviewers assumed that the study's analysts collected accurate information about the plants' piping systems and instrument layouts.

Sandia's final report findings differed slightly from its draft report as a result of further analysis, NRC staff review of the draft, and meetings with the study's analysts to clarify certain issues. The study's authors reanalyzed some aspects of the study following Sandia's draft report, and Sandia's analysts incorporated the revised analyses into their final report.

Although the NRC has not yet published a summary of its review of the study, NRC's testimony at the Indian Point hearings contained some information on the results of NRC's reviews of the study. However, the testimony primarily presented NRC's own estimates of the risks at the Indian Point plants and did not represent a full review of the study. While NRC developed and presented separate risk estimates of the Indian Point plants at the hearings, according to NRC, its risk estimates could not be directly compared to the study's risk estimates because of differences in methodologies and data interpretations. Therefore, the summary of the peer review findings in the next section concentrates on comments and criticisms of the Indian Point study and not on a comparison between the study's and NRC's risk estimates of the Indian Point plants.

Advisory Committee on Reactor Safeguards

The ACRS, a statutory body of advisors to the NRC, hired seven consultants to do a limited reading and review of the Indian Point study to assess the general validity and/or amount of uncertainty in the study. The ACRS consultants reviewed those portions of the study that they felt deserved attention. The ACRS received the consultants' reports last summer and fall and conducted a meeting on the review of the Indian Point study last November.

For the most part, the ACRS consultants performed cursory reviews of the study. For example, the Argonne National Laboratory review, the most extensive of these reviews, was limited to only a few sections of the study. Also, the Argonne reviewers mostly concentrated on plant design differences between the Zion (near Chicago, Il.) and the Indian Point nuclear powerplants. Argonne had already reviewed the Zion study, a sister study to the Indian Point study which was also prepared by Pickard, Lowe, and Garrick, Inc.

Intervenors

Some of the intervenors to the Indian Point hearings, specifically the Union of Concerned Scientists, the New York Public Interest Research Group, the Friends of the Earth, and the Audubon Society, will have to present testimony on their evaluation of the risks of the Indian Point plants. According to the intervenors, to fulfill this task, they plan to question the validity of risk assessments and identify a few specific flaws in the Indian Point study because a complete and independent PRA is beyond their resources. Although the intervenors have presented some testimony, the extent and results of their criticisms of the study will not become fully known until the hearings are over.

Peer review is incomplete and limited

The Sandia National Laboratories have stated that "A review of PRA is not complete unless the information and analysis which comprises each task is examined." To date, no single review or compendium of reviews of the Indian Point study fulfills this criterion of a complete review. Peer review of the Indian Point study is incomplete and limited because the study is complex and innovative, and portions of the study are inadequately documented.

As discussed in an earlier section, level-three PRAs--like the Indian Point study--are diverse in scope and require input from many different technical specialists. Consequently, most reviewers of the Indian Point study focused on specific sections of the study, noting that their reviews were "limited" in scope and depth. For example, NRC consultants reviewing the study's analysis of fires examined how the study "identified, analyzed, and quantified" fires because they could not possibly check all the details of the study's fire analysis without redoing the analysis from scratch.

Peer review of the Indian Point study is also limited because, according to an NRC consultant who reviewed the study, the engineering community has not had enough time to become familiar with and use the innovative portions of the study--like the external event analyses--which extended the state of the art. The study's earthquake analysis, for example, is pioneering and represents one of the few times that the methodology has been used for full-scale reactor risk assessments.

Further, although the Indian Point study is voluminous, reviewers frequently criticize the study for inadequate documentation and insufficient justification. Both the human reliability and accident sequence analyses of the study were specifically criticized for inadequate documentation. For example, a Sandia analyst found it "difficult to impossible" to evaluate the estimates of some of the human error probabilities because of the study's "lack of documentation and the difficult to follow format." Sandia analysts also commented that the Indian Point study's accident sequence analysis was difficult to follow because of "unclear or incomplete description of events or the modeling of them."

Reviewers also cite the study for insufficient justification of the analysts' assumptions and judgments. As previously discussed, this has frequently been cited as a problem in validating the results of PRAs. For example, reviewers from the Argonne National Laboratory questioned the study analysts' selection of numerical values used to determine the frequency of large fires in the cable spreading rooms of the plants because of the "very minimal justification" in the study.

Given the status and limitations of the peer review of the Indian Point study, some reviewers have called for additional review and validation of the study to more closely evaluate the methodological limitations and differences in engineering judgment.

PEER REVIEW FINDINGS OF THE INDIAN POINT STUDY

Many of the preliminary peer review comments and criticisms of the Indian Point Probabilistic Safety Study indicate that, although the study represents and even extends the state of the art in PRA methodology and knowledge, it contains all of the significant limitations inherent in PRA. That is, it suffers from the same limitations in data, modeling, and completeness as all PRAs.

Peer reviewers generally found that the uncertainties in the study's data, models, and analyses are greater than the study's analysts estimated. While reviewers have identified several significant errors, omissions, and critical judgments in the study that may affect the study's risk estimates, the reviewers emphasized that the large overall uncertainties associated with the study render precise risk estimates unreliable.

Major peer review finding

The Indian Point study may have underestimated or overestimated the actual risk of the Indian Point plants because the large overall uncertainties associated with the study render the study's precise risk estimates unreliable. In addition, the study's treatment of uncertainties in probability estimates is controversial within the field of PRA. Consequently, peer reviewers emphasized that their numerical differences with the study are not significant considering the large overall uncertainties involved.

Uncertainties greater than the study estimated

Peer reviewers have observed that the study's uncertainties are probably larger than the study's analysts estimated, but the reviewers do not know how large the uncertainties really are. In other words, according to the peer reviewers, the study's broad statements of uncertainties, or confidence levels, should be even broader than the study stated, especially for the external event analysis.

As explained in an earlier section, confidence levels express the analysts' degree of confidence that a frequency or risk estimate lies within a certain range, or bound, of estimates. Confidence bounds establish the parameters of estimates based on available data and knowledge. Therefore, confidence levels are the analysts' statements of their certainty (or uncertainty) about an estimate.

According to peer reviews, the Indian Point study's confidence bounds are too narrow because the uncertainties in the study's data, models, and analyses are greater than the study analysts estimated. Consequently, NRC reviewers concluded that

> "* * * it is quite plausible * * * that the true risks posed by severe reactor accidents at Indian Point units 2 and 3 might lie outside the range suggested by the uncertainty calculations in the [study], either toward higher or lower risks."

Further, because the study's uncertainty bands should be much wider than the study's analysts estimated, the peer reviewers' numerical differences with the study's estimates are not considered significant. The following examples illustrate this point.

The NRC staff revised the study's plant damage state likelihood estimates following peer reviews of the study's accident sequence analysis. (Plant damage states are groups of accident sequences that cause similar physical responses in a nuclear powerplant.) The combined total of these estimates represents the estimated probability of core melt from all sources.

The following table compares the study's mean and 90 percent confidence level estimates of core melt frequency for the Indian Point plants with NRC's point or "best" estimate. As explained earlier, the study's 90 percent confidence level represents the analysts' upper bound on the range of uncertainty assigned to the estimate. For example the study's analysts were

90 percent confident that the frequency of core melt at Indian Point unit 2 is not more than once in 1,000 years of reactor operation. The analysts' best estimate in this case is represented by the mean of not more than once in 2,100 years.

Comparison of Total Core Melt Frequency From All Accident Scenarios

	Indian Point study (note a)		NRC Estimate (note b)
	Mean	90% confidence level	Point estimate
	---------(in reactor years of operations)-------		
IP-2	1 in 2,100	1 in 1,000	1 in 1,000
IP-3	1 in 5,300	1 in 1,800	1 in 1,500

a/"The Indian Point Probabilistic Safety Study," Spring 1982.

b/Docket Nos. 50-247-SP and 50-286-SP, Direct Testimony of NRC on Commission Question One, 1/24/83; "Before Fix".

As the table above demonstrates, NRC's "best" estimates of core melt are about two times more frequent for unit 2 and about three and a half times more frequent for unit 3 than the study's mean, or best, estimates. Further, NRC's "best" estimate of core melt frequency for unit 3 exceeds the study's 90 percent confidence level. However, NRC believes that these numerical differences are not significant because the study's analysts understated the range of uncertainty associated with these estimates. In other words, according to NRC, its best estimates of core melt frequencies for the Indian Point plants would fall within the uncertainty bounds if the uncertainty bounds had been properly stated and thus do not substantially differ from the study's estimates.

Even when the peer reviewers' numerical estimates seem to differ substantially from the study's estimates, the large uncertainties associated with these estimates overshadow the differences. For example, reviewers at the Sandia National Laboratories concluded that the study analysts underestimated the frequency of core melt due to hurricanes at Indian Point 2 because of "questionable assumptions made about the hurricane hazard." The Sandia reviewers estimated that hurricane-initiated core melt at Indian Point 2 is 20 times more frequent than the study analysts estimated. However, other NRC reviewers believed that this difference was "modest in an absolute sense" because the uncertainties in the analysis are very large.

Treatment of uncertainty controversial

Although the NRC and other peer reviewers of the Indian Point study have commended the study's analysts for their pioneering efforts in treating uncertainties throughout their risk calculations, reviewers observed that this treatment is controversial within the field of PRA. While the NRC believes that the study's treatment of uncertainties is at least reasonable for bounding internally initiated accident sequence estimates, reviewers generally questioned the subjectivity and consistency of the study's treatment of uncertainties.

As previously discussed in the study's methodology section of this report, the study's analysts used a Bayesian methodology to quantify their estimates of the uncertainties throughout their risk calculations. Using the Bayesian approach, the study's analysts were able to quantitatively organize and introduce information, beyond that contained in the observed data, into their analyses. This additional information, based on the analysts' experience, engineering judgment, and expert opinion, contributed to the analysts' estimates of the range of potential error in their risk estimates.

Basically, the study's analysts probabilistically developed prior beliefs--"prior distributions"--about rates and probabilities based on published information and their own state of knowledge. Then the analysts modified their prior beliefs using available Indian Point data to calculate a probability distribution representing their "posterior" or revised beliefs about the frequencies and consequences of various accidents.

Employing the Bayesian methodology in nuclear powerplant risk assessments is relatively new and still controversial within the field of PRA. Experts differ as to whether an analyst can precisely translate his or her knowledge and beliefs to probability. Further, experts do not always share the same beliefs and consequently disagree on the probabilities. As one indication of this disagreement, the PRA Procedures Guide cautioned the Bayesian analyst that it will be "difficult to convince the community at large to adopt his degree of belief as their own." This controversy extends to the peer review comments of the Indian Point study.

Intervenors to the Indian Point hearings maintained that the study's use of the Bayesian technique has not been fully evaluated and that the analysts' judgments are arbitrary and cannot be trusted except where there are large amounts of plant-specific data present. Further, an intervenor argued that "one should not rely on the analysis contained in the [study]" until a complete reassessment of the data on Indian Point units 2 and

3 is done in a manner which recognizes that PRA analysts cannot make precise estimates of probabilities.

In testimony before the Indian Point hearings, NRC acknowledged that although it sees some merit in the study's use of engineering judgment to arrive at a comprehensive treatment of uncertainties, it is, nonetheless, subjective. Also, the NRC staff did not attempt to undertake a subjective estimate of this kind. According to NRC, the staff did not attempt to formally calculate the uncertainties in their own risk calculations for the Indian Point plants "principally because there are many sources of uncertainty, such as modeling approximations and completeness issues for which the uncertainty cannot be mathematically derived."

In an attempt to evaluate the study's use of the Bayesian technique, Sandia reviewers compared the confidence levels associated with several internally initiated accident sequence estimates derived from the study's Bayesian methodology with those derived from an adaptation of classical statistics. According to NRC, Sandia's results were not significantly different from those in the study. From this NRC inferred that the choice of statistical method is not a large source of uncertainty or potential error in most of the accident sequence likelihood estimates.

However, the Sandia reviewers compared only the confidence bounds of internally initiated accident sequences. Generally, one would expect that the difference in results between statistical methods would naturally be less when more data is available, which is the case for this portion of the risk assessment. For example, Sandia reviewers found a few cases where the confidence bound estimates were quite sensitive to the statistical method chosen when Indian Point data was not available or used.

Moreover, the Sandia analysts did not attempt to estimate the confidence bounds for externally initiated events or for combined internally and externally initiated plant damage state frequencies because of the "paucity of data and immaturity of the methodology." Other peer reviewers have also commented that the state of the art is not yet developed enough to estimate confidence bounds for external events.

In addition, reviewers from the Sandia and Argonne National Laboratories questioned the study's consistency in treating uncertainties associated with estimates from existing data sources. The study's analysts modified the confidence bounds reported in the Reactor Safety Study (WASH 1400) and the NRC Human Reliability Handbook to express greater uncertainty in the component failure rates and human error probabilities obtained from these sources. According to the study's analysts, this approach allowed them to express greater uncertainty about the error rates and probabilities than these generic sources of data recommended. The study's analysts considered this a conservative treatment of uncertainty.

However, the reviewers noted that the study's analysts inconsistently modified the confidence bounds from these data sources. For example, in their comments on the Zion Probabilistic Safety Study, which they said held true for the Indian Point study, the Argonne reviewers observed that the study's analysts did not expand the confidence bounds for human error probabilities as much as they expanded the confidence bounds for component failure rates. According to the Argonne reviewers, this "appears to be a somewhat arbitrary decision" because "our ignorance of human error rates exceeds our ignorance of equipment failure rates." Further, the reviewers noted that the uncertainty estimates can be quite sensitive to what appear to be minor differences in assumptions.

While both the Sandia and Argonne reviewers acknowledged that the Bayesian approach allows the study's analysts to assign whatever confidence bounds they believe to represent their state of knowledge, the reviewers called for more justification to support these decisions.

Other important peer review findings

While many peer reviewers praised the Indian Point study for being the most complete, credible, and readable PRA produced to date, the reviewers have a number of general and technical differences with the study, some of which they believed to be significant. Some of the peer review comments and criticisms of the Indian Point study include:

--The study does not accurately represent the Indian Point plants.

--The study omits several potential initiating events and possible accident scenarios which may be significant.

--The study's external event analysis is limited because the state of the art is still developing.

--The study's containment and consequence analyses and models may be limited.

The basic reasons for these differences are discussed below. While peer reviewers had many other differences with and suggestions for the study's authors, the peer review comments and criticisms highlighted here illustrate the type and possible significance of the limitations in the Indian Point study.

The study does not accurately represent the Indian Point plants

According to Sandia analysts and NRC officials, the study does not accurately represent the design of the Indian Point plants at the time the study was done. During the study, the licensees identified five potential plant design problems and decided to implement corrective modifications. The study reflects the design of the Indian Point plants plus these modifications. While reviewing the study, the Sandia analysts learned that the licensees planned to defer one of these modifications, which would have helped prevent an "anticipated transient without scram".

An anticipated transient without scram occurs when a nuclear powerplant experiences a severe abnormal operating condition and the plant's safety control systems fail to shut down (also known as scram) the nuclear reaction. According to NRC, the loss of off-site power and the loss of feedwater are two examples of abnormal operating conditions. Initiating a scram is an important safety measure whereby control rods are inserted into the reactor core to shut down the nuclear reaction.

Following the Sandia review, the study's analysts revised the study's anticipated transient without scram event tree to reflect the deferred modification. According to Sandia's reviewers, the revised analysis revealed that without implementing the modification, the anticipated transient without scram core melt frequency for Indian Point 2 would be about 17 times higher than the study's analysts originally estimated, and about 23 times higher for Indian Point 3. The Sandia analysts factored these revised frequencies into their revised plant damage state frequencies.

The study omits several potential initiating events and possible accident scenarios which may be significant

As previously explained, the issue of completeness is a major source of uncertainty in PRA because PRA analysts may not completely identify all possible accident sequences or systems failures. Although peer reviewers generally considered the Indian Point study relatively complete and consistent with ongoing PRAs, they observed that the study does omit several potential initiating events and accident sequences, some of which the reviewers believed may be significant to the risks of the Indian Point plants. Some of the more important initiating events and accident sequences which the study omitted are discussed below.

Sabotage--Like all PRAs conducted to date, the Indian Point study did not consider sabotage in its analysis of possible accident scenarios at the Indian Point plants. According to NRC, this event cannot be modeled within the current state of the art of PRA because the likelihood of sabotage cannot be estimated.

Intervenors to the Indian Point hearings argued that omitting sabotage introduces "substantial uncertainty in the estimates of risk" because the Indian Point plants "may be a more visible and attractive target for sabotage due to the proximity of the plant [sic] to New York City." However, the significance of this omission on the risks of the Indian Point plants cannot be determined from the peer review of the study.

Pressurized thermal shock--Neither the Indian Point study's authors nor their peer reviewers explicitly evaluated pressurized thermal shock. This potential initiating event occurs when a reactor vessel, damaged by long-term exposure to radiation, is rapidly cooled while under pressure. This event could rupture a reactor vessel and possibly initiate a core meltdown. Currently, the NRC and the nuclear industry are investigating the problems of pressurized thermal shock. Although their research is incomplete, NRC has predicted that the Indian Point frequencies of reactor vessel failure due to pressurized thermal shock are acceptable.

Loss of component cooling water--Analysts at the Sandia National Laboratories reviewing the study determined that the study's authors did not analyze the probability of a loss of component cooling water due to a pipe break initiating an accident. The component cooling water system of a nuclear powerplant cools several important pieces of equipment during all phases of a plant's operation--including safety pumps that must operate to prevent a core from melting. A large pipe break within the system, disrupting the flow of cooling water to these vital pieces of equipment, could lead to a core melt. According to the Sandia analysts, this potential initiating event, which the study omitted, is actually the leading contributor to core melt at Indian Point unit 3. On the other hand, the Sandia analysts also recognized that quantifying pipe break estimates entails large uncertainties.

Steam generator tube ruptures--Also during their review, Sandia's analysts noted that the study's steam generator tube rupture event tree was incomplete because it did not model a rupture with a stuck-open secondary safety valve. However, the probability of this accident is fairly low.

In pressurized water reactors, such as the Indian Point plants, the steam generator transfers heat from the reactor's closed radioactive cooling water system, or primary system, to a separate nonradioactive water system called the secondary system. The heated water from the primary system passes through thousands of tubes in the steam generator, turning water in the surrounding secondary system to steam. This in turn drives the turbines that produce electricity. Steam generator tube ruptures are potentially high-risk accidents because if the tubes rupture, water in the secondary system would become radioactive and radiation could be released directly into the atmosphere.

NRC estimated that the probability of a steam generator tube rupture initiating a core meltdown at the Indian Point plants is small, roughly 1 in 250,000 reactor years of operation. The study's authors are redoing the study's steam generator tube rupture event tree analysis according to the Sandia analysts.

Systems interaction accidents--The Indian Point study, like other risk assessments, may not have completely analyzed systems interaction accidents. According to NRC, it is widely recognized that these accidents are subtle and hard to anticipate. For example, an intervenor to the Indian Point hearing argued that these types of accidents involve the "interaction of small failures in independent subsystems that can mysteriously or unexpectedly become linked together." The intervenor claimed that the Three Mile Island accident is an example of a systems interaction accident. However, the significance of this omission is unknown. The Power Authority of the State of New York is evaluating systems interactions at the unit 3 plant. The licensee's study, expected to be finished this year, may indicate how well the Indian Point study modeled these interactions.

Other omissions--The study's analysts also omitted several initiating events and accident scenarios which they and in some instances the peer reviewers found to be insignificant to the risks of the Indian Point plants. For example, several reviewers criticized the study for not explicitly considering the possibility of a Ramapo fault zone--a suspected geological fault zone near the Indian Point region--in its seismic analysis. However, Sandia analysts investigating the possible impact of this suspected fault zone concluded, like the study analysts, that insufficient evidence exists to consider the Ramapo fault zone as an active earthquake generating source and that this omission in the study did not significantly affect the study's results.

The study's "other" category--The study's "other" category did not provide for the study's significant omissions, according to an analyst at the Sandia National Laboratories. The study's analysts created an "other" category to provide for unknown or unimagined events, establishing what they believed was a conservatism in the study. However, the study's analysts assigned low probabilities to the category because they believed that the "other" events had not yet occurred, for if they had, the study's analysts would have included them on their list of possible accident scenarios. The study's low probability "other" category therefore did not provide for higher probability events which were omitted, like the loss of component cooling water due to a pipe break.

External event analyses limited

In general, although peer reviewers commended the study's analysts for attempting to analyze the contribution to risk from externally initiated events--such as earthquakes, hurricanes, fires, and floods--they noted that PRA "external event analysis * * * is still in its infancy." Consequently, peer reviewers observed that large uncertainties in the study's external event analysis due to inadequate data, immature methodology, and considerable controversial judgment preclude quantitative conclusions about the size of the risks of the Indian Point plants. In other words, according to peer reviewers, the study's conclusion that external events are the dominant contributors to risk at the plants cannot be reliably quantified in a precise manner. While the study's earthquake analysis, for example, is at the forefront of the state of the art, NRC consultants reviewing the study noted that it is quite controversial because it is so new and involves a good deal of judgment.

The study's external event analysis is limited because of difficulties in estimating the likelihood of a major event and the corresponding effect of these events on the plants. According to an NRC consultant testifying at the Indian Point hearing,

> "the likelihood of a major event (* * * a very large earthquake or an extreme flood) is often neither known from the historical record nor reliably inferred from analysis based on extrapolations from that record."

Further, estimating "the effect of these events on plant components, systems and functions is in some cases not well understood." Estimates of the plants' fragility to severe external events were also based on incomplete data and approximate analysis and therefore relied heavily on the judgment of the study's analysts.

Because of the limitations in the data and analyses, peer reviewers of the study believed that the uncertainties in the study's external event methodology are larger than the study's analysts estimated. An NRC consultant testifying before the Indian Point hearings stated that "the [large] uncertainties quoted in the [study] may be understated, especially for the seismic and high-wind risks." Other reviewers have noted that the uncertainties in the study's release frequencies from externally initiated accidents are very large and point estimates could be off by many orders of magnitude. In other words, the actual externally initiated core melt frequencies may be several times higher or lower than the stated estimates, but the true frequency is uncertain.

External event analysis is still in the developmental stage, according to the study's peer reviewers. Currently, external event methodology is not well enough developed or data well enough known, or both, to reliably quantify the resulting risks. In their final report the Sandia reviewers concluded that "the [study's] external event data and the methodological models, as well as the alternate data and models used in this review, are somewhat simplistic." While this may not necessarily invalidate the study's external event analysis, it nonetheless demonstrates the significance of the large overall uncertainties associated with the study's external event estimates.

Limitations in the containment and consequence models

Peer reviewers have criticized the study's containment and consequence models for several reasons. Some of the peer review criticisms of these models are discussed below.

Containment model limitations--While the study's analysts determined that the Indian Point plants' containment structures reduce the risks from accidents at the plants, reviewers from the NRC found that "* * * there appears to be greater uncertainty in the likelihood and character of containment failure modes than the [study] takes into account." NRC staff reviewing the study observed that the study's description and analysis of the core melt process represented only one among a variety of possible courses that accidents might take.

The uncertainties associated with containment analysis--estimating how a core meltdown progresses--are very large. One ACRS consultant, reviewing the study, concluded that the study's risk estimates may be underestimated or overestimated by about a factor of 10. In other words, the true risk may be 10 times higher or lower than the study's estimates.

Also, several reviewers criticized the study for not analyzing environmentally accelerated equipment failure. The reviewers claimed that the study's analysts did not demonstrate that the plants' containment heat removal systems would function in an adverse core melt environment. That is, some peer reviewers questioned whether the Indian Point plants' containment spray and fan systems, which protect the containment from overpressure during a core melt accident, would indeed survive such an accident.

The Indian Point plant containments have two spray systems--the containment spray injection system and the containment spray recirculation system--and a fan cooler system. The spray injection system draws water from the refueling water storage tank. When water in the tank is depleted, the injection system pumps shut down. Then the recirculation spray system recirculates water drawn from the containment sump.

According to peer reviewers, the study's analysts gave more credit to the containment spray systems than may be justified. The study's analysts did not take credit for operation of the spray recirculation system partly because this system might fail due to the physical processes associated with the core meltdown. However, the study's analysts also assumed that the Indian Point operators would refill the refueling water storage tank and thereby use the spray injection system throughout an accident. NRC and Sandia reviewers questioned the validity of this assumption because the Indian Point emergency procedures do not cover refilling the storage tank to operate the injection system during recirculation. Assuming that the spray injection system is not available during the recirculation phase, Sandia analysts calculated that the NRC-defined plant damage state, "Late Core Melt Without Containment Cooling," would occur more frequently than the study's analysts estimated.

The Sandia analysts also investigated the effect of failure of the Indian Point containment spray and fan cooler systems on the study's plant damage state estimates. For their calculations, the Sandia analysts postulated that the fan cooler system would fail during a core melt and that the containment spray injection system would not be available during the recirculation phase as discussed above. (Sandia's analysts also assumed, like the study's analysts, that the recirculation spray system probably would not be available. It should be noted that the Sandia analysts did not resolve whether the Indian Point fan cooler system would work during an accident because Sandia and NRC are currently researching this potential problem.) According to the Sandia analysis, given these assumptions, one plant damage state--Late Core Melt Without Containment Cooling--would occur much more frequently than the study's analysts estimated. However, NRC has determined that the increase in risk, due to containment spray and fan cooler systems failure, is slight. Determining the probability of the fan cooler system failure and the resulting impact on the risks of the Indian Point plants needs further study.

Consequence model limitations--The study's consequence model may have underestimated the health effect consequences of possible accidents because it may not have considered the impact of large external events, such as earthquakes, hurricanes, etc., on the evacuation of the public. However, the certainty and significance of this limitation is unknown because the study's consequence model has not been available for complete peer review.

Although the study's analysts determined that external events, especially earthquakes, wind, and fires, are among the dominant contributors to risk at the Indian Point plants, the study's consequence model did not account for the likely effects that these events would have on the evacuation of the public in case of a possible accident. Several peer reviewers of the study, including an ACRS consultant and an intervenor to the Indian Point hearings, suggested that external events severe enough to initiate an accident would also have an adverse effect on the necessary evacuation of the public. In their testimony before the Indian Point hearings, the NRC staff acknowledged that the study's consequence model probably underestimated the health effect consequences because of this limitation.

Peer reviewers were unable to either confirm the existence or estimate the significance of this limitation because the study's consequence model is a proprietary and complex computer program that is not available in the open literature. The consequences of possible accident scenarios were estimated in the study using the CRACIT computer program. CRACIT, which stands for Calculation of Reactor Accident Consequences Including Trajectories, was developed by Pickard, Lowe, and Garrick, Inc. CRACIT is an extensive modification of the NRC's CRAC, Calculation of Reactor Accident Consequences, model.

CRACIT has been compared to CRAC and other consequence computer models in an international research project conducted under the auspices of the Committee on the Safety of Nuclear Installations of the Nuclear Energy Agency, Organization of Economic Cooperation and Development. The detailed results of this research are to be published sometime this year. In preliminary reports, CRACIT compared "reasonably well" with NRC's CRAC model. However, according to the NRC, the CRACIT model handles evacuation in a complicated manner very different from CRAC, and no comparisons are available involving the evacuation models of both programs.

Summary

Although full peer review of the Indian Point study is incomplete and limited, reviewers generally concluded that the study's precise risk estimates are unreliable because the uncertainties are greater than the analysts estimated. The uncertainties in the study also prevented the peer reviewers from precisely quantifying risks. For example, even after revising the study's core melt frequencies, achieving what they believed was a state-of-the-art level of completeness, NRC reviewers concluded that they could not confirm that their revised core melt frequencies were correct in an absolute sense because of the uncertainties associated with completeness, data, and modeling. Finally, as noted earlier, the results of a PRA depend largely on the judgments and assumptions of the analysts conducting an assessment. Peer review differences with the Indian Point study indicated that these judgments and assumptions can vary significantly among PRA experts.

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