POWER AUTHORITY OF THE STATE OF NEW YORK
10 COLUMBUS CIRCLE, NEW YORK, N.Y. 10019

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
4 IRVING PLACE, NEW YORK, N.Y. 10003

March 5, 1982

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

SUBJECT: Indian Point Unit 2
Docket No. 50-247

Indian Point Unit 3
Docket No. 50-286

Indian Point Probabilistic Safety Study

Dear Mr. Denton:

Enclosed are fifty (50) copies of the twelve (12) volume report entitled, "Indian Point Probabilistic Safety Study". This study was undertaken by the Power Authority of the State of New York and the Consolidated Edison Company of New York, Inc. and was performed by Pickard, Lowe & Garrick, Inc., Westinghouse Electric Corporation and Fauske & Associates, Inc. Also enclosed are fifty (50) copies of an "Overview and Highlights of the Indian Point Probabilistic Safety Study".

This rigorous and comprehensive study presents on a probabilistic basis, the risks associated with the operation of Indian Point Units 2 and 3. During the course of performing this plant evaluation we determined areas where cost effective risk reductions could be attained and have taken steps to implement these improvements. These plant improvements which are being implemented are listed as items (1) through (6) in Attachment A to this letter and have been factored into the study model. In addition, item (7) of Attachment A identifies a Confirmatory Order commitment and items (8) and (9) present independent prior regulatory commitments which have also been incorporated into the study model.

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Employing the plant models developed in the course of the study, we are also evaluating features that may aid in the reduction of risk for their safety significance. A further report on these features will be prepared in the near future.

We look forward to discussing the study with you and your staff.

Very truly yours,

J. P. Bayne
Senior Vice President
Power Authority of the State
of New York

William P Bennett
for John D. O'Toole
Vice President
Consolidated Edison Company
of New York, Inc.

Enclosure

ATTACHMENT A

(1) Implementation of a refueling interval surveillance test to verify disc integrity for RHR valves MOV-730 and MOV-731.

(2) Implementation of a refueling interval surveillance test to verify disc to stem integrity for spray valve 869A (such verification is already accomplished for redundant valve 869B in the course of performing normal refueling operations).

(3) Implementation of changes to refueling interval surveillance test to clarify test method and data recording for accumulator discharge valve flow verification check.

(4) Implementation of procedures to periodically verify that: (a) the service water pumps designated for essential header service are in fact aligned to the required essential services (i.e., field valve alignment check), and (b) the position of the control room service water system mode selector switch is such that the correct set of service water pumps is selected for automatic essential header service.

(5) System modification, procedural change or verification testing to ensure that sufficient backpressure will be maintained in the service water system to prevent service water pump overload for cases where only one service water pump is operating with the system in accident configuration.

(6) Rearrangement of diesel generator fuel oil transfer pump power supplies such that the primary transfer pump for each diesel is powered from one of that diesel's electrical buses (Indian Point 2 only).

(7) Confirmatory Order Item C.4: Implementation of plant modifications for mitigation of ATWS.

(8) Replacement of manual isolation valves with motor operated isolation valves in certain of the fan cooler service water discharge lines (Indian Point 2 only).

(9) Implementation of masonry wall upgrading modifications for station batteries in response to IE Bulletin 80-11.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 8, 1982

MEMORANDUM FOR: DISTRIBUTION

FROM: Thomas M. Novak, Assistant Director
for Operating Reactors
Division of Licensing
Office of Nuclear Reactor Regulation

SUBJECT: DISTRIBUTION OF INDIAN POINT PROBABILISTIC RISK ASSESSMENT (PRA)

The long awaited Indian Point PRA has arrived. The report is copyrighted so a controlled distribution is being made. However, sufficient copies are available to permit those who need them to be supplied. Please contact this office if additional copies are required.

T M Novak

Thomas M. Novak, Assistant Director
for Operating Reactors
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
Indian Point PRA

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TELECON NOTE

Docket No. 247/286 SP

MEMORANDUM FOR: File

FROM: Hannon

TELECON: Bill Pennett Con Ed (212) 460-2533

Time and Date: 3:10 3/8/82

Subject: IP PRA

Purpose: Copy Right Provisions

Bill said that he had checked with Lawyers and they had no problem with us microfilming the PRA and sending it to the PDR.

Special Instructions: Inform Steve Scott

I already have copy 33

H

Project Manager
Operating Reactors Branch #2
Division of Licensing

INDIAN POINT PRA

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Indian Point Probabilistic Safety Study

Overview and Highlights

Power Authority of the State of New York

Consolidated Edison Company of New York, Inc.

1982

INTRODUCTION

In the past decade, society has grown increasingly aware of the direct and indirect risks that can accompany technological advancements. This awareness has led to new laws, new regulations, and new scientific methods for measuring technological risks. These new techniques attempt to quantify the two elements that make up "risks": the likelihood of the damage occurring and the magnitude of any potential damage—or in risk assessment terms, the "probabilities" and the "consequences." These two components of risk—probabilities and consequences—cannot be separated; both are essential to decisions about benefits and the relative merits of alternative courses of action. By calculating both components of technological risk as precisely as possible, society can more knowledgeably compare the relative risks of competing technologies. Then, the risk of a technology in comparison to its benefits can be examined.

The energy production industries, particularly nuclear power, have been at the forefront of these advances in risk assessment. Nuclear power has adopted an advanced, sophisticated approach in detecting and measuring risks which has furthered our understanding of nuclear power plant safety. When all other forms of energy are similarly evaluated, a truer picture of the tradeoffs in energy decisions will emerge.

The purpose of this overview is to present some of the highlights of an extensive safety study performed on Indian Point Units 2 & 3. Since the study is a state-of-the-art investigation using sophisticated scientific tools, this overview provides some perspective for the general readership. It includes a discussion of nuclear power plant safety features and reactor safety analyses. Then, it describes a methodology called "probabilistic risk assessment" (PRA) followed by highlights of the Indian Point Probabilistic Safety Study.

NUCLEAR POWER PLANT SAFETY

In decisions to license, build, and operate all nuclear power plants, the issue of safety dominates. Operators of nuclear power plants must demonstrate to the Nuclear Regulatory Commission (NRC)—the independent Federal agency responsible for licensing and regulating nuclear facilities—that each plant is designed and constructed with adequate safety features. Most of these safety features have one overall objective: to prevent or minimize accidents which can result in offsite radiation exposure or release of radioactive material from the plant.

Reactor Design Features

Several physical barriers to prevent radioactive materials from escaping to the environment are designed into every nuclear power plant. They include:

- *Fuel Rods*. The tubes, or fuel rods which hold uranium fuel pellets, are made of a strong alloy called zircaloy which helps prevent the contained solids and gases from spreading through the reactor coolant system.
- *Reactor Vessel*. Surrounding the core of fuel rods is a reactor vessel some 8 inches thick manufactured of alloy steel to the most rigorous standards and lined with stainless steel.
- *Containment Building*. The reactor and its coolant system are surrounded by a massive concrete and steel building which is specially designed to prevent radioactive materials from reaching the environment in the event that piping systems should leak or break. The concrete in the containment is typically some 3 feet thick and lined with welded steel plate.

In addition to these physical safety barriers, nuclear power plants are designed and built with multiple and diverse safety systems. Outside the plant and at the site boundary, sensitive monitoring and surveillance instruments are installed to detect radiation releases.

Scientists are able to detect and measure radiation even in minute amounts better than virtually any other substance known to man. According to the Committee on the Biological Effects of Ionizing Radiation (BEIR) of the National Academy of Sciences, the average American receives about 100 millirems (a millirem—or one-thousandth of a rem—is a standard unit of radiation dose measurement) each year from background radiation. Natural background radiation comes from the sun, minerals in the earth, and other naturally radioactive elements in the air and in our food. Nuclear power plants and related activities contribute on the average only 0.3 millirems each year to natural background radiation levels under normal operating conditions.

Nonetheless, an accidental release of radioactive materials to the environment with higher exposure levels remains a remote possibility. Because this potential impact upon public health and safety exists at all, emergency plans have been developed and studies are done to continually improve the safety systems in nuclear power plants.

Reactor Safety Studies

Safety analyses are performed for every nuclear power plant before it begins operating. These analyses determine whether the plant owners and operators have taken the appropriate precautions for safe operation of the plants. The techniques for conducting these studies have been refined over the years as the nuclear industry gains more experience. These refinements and improvements are made possible by a growing body of information about the performance of the designed safety features, the reliability of components, and the adequacy of safety margins for different systems and components. Further refinements are made possible by advances in computer technology—the ability to process and analyze large quantities of information to the smallest detail—and in the analytical techniques used by scientists to unravel complicated series of events and to estimate the impacts of those events.

From the early days of the nuclear industry—more than 25 years ago and 1,500 operating years of experience worldwide—safety thinking centered around the multiple-barrier approach: the physical layers of protection (such as the containment) and the series of backup safety systems (such as emergency core cooling) in case the primary system should malfunction. This safety concept is termed "defense in depth."

Observing "defense in depth" guidelines led to design requirements for nuclear plants that included hypothetical problems called design basis accidents and maximum hypothetical accidents. Engineers had to consider what damage could be caused by, for example, a loss of coolant to the reactor vessel, an earthquake, an airplane crash, a fire or pump failure. They then designed the different parts of the plant to withstand such accidents if these events were to occur.

Since physical simulation of each hypothetical accident is not feasible, additional calculations were made for the most serious possible damages to the plant. These "upper bounding" calculations helped ensure that the best engineering judgments about safety—for example, how much stress a piping system could take in the event of an earthquake—incorporated extra safety margins.

The "defense in depth" philosophy has served the cause of nuclear safety well. Carried to an extreme, however, it can be counterproductive. The introduction of unnecessary complexity could cause a net reduction in safety instead of the expected increase in levels of protection.

Therefore, with the accumulation of a substantial body of nuclear operating experience, scientists determined that tools other than "upper bounding" calculations were needed to make more accurate and realistic safety decisions about nuclear power plants.

Leaders in the field began to look at some of the new scientific tools that other industries—aerospace and defense, for example—were developing to deal with their own questions about safety, performance, and the risks involved. These tools seemed to suit many of the same issues that the nuclear industry confronted. Yet much more work needed to be done to achieve the accuracy and realism that the nuclear industry desired. Experts from a number of different fields pooled their knowledge to address the problem: how can we systematically evaluate even the most improbable accidents to determine the risk they could present to public health and welfare?

The Reactor Safety Study, commissioned by the NRC and directed by Dr. Norman C. Rasmussen of the Massachusetts Institute of Technology, was the first comprehensive study of the accident-related risk of nuclear power plants. Published in October 1975 after three years of work, it was the first attempt to quantify the risks resulting from nuclear power plant operation. It enabled a systematic classification of accidents according to their possible frequency and the possible consequences that could result.

The benchmark Reactor Safety Study report was thoroughly reviewed and critiqued by numerous groups in the years following its publication. The Lewis Report, an evaluation performed for the NRC by the Risk Assessment Review Group, contains both praise and criticism for the original study.

According to the Lewis Committee, the Reactor Safety Study was a significant improvement over earlier attempts to calculate the risks of nuclear power. It introduced a workable accident classification scheme and applied the rules of mathematical probability theory in order to quantify risks. It evolved "event tree" and "fault tree" procedures—described later in this report—to quantify the frequencies with which accidents could happen. It also examined a broader range of potential health effects: in addition to expressing risk in terms of injuries and fatalities that could occur immediately following an accident, the Reactor Safety Study considered the delayed effects of an accident by estimating latent fatalities and cancers. The Lewis Committee also criticized the Reactor Safety Study primarily for its lack of an adequate data base on which to perform some of the analyses and the way uncertainties in the results were portrayed.

Advances in reactor safety analysis since the Reactor Safety Study include:

- More extensive operating data and improved methods for handling data, including the treatment of uncertainty.
- Better documentation and models for systems to reflect the interaction of reactor operators and accident conditions.
- More comprehensive treatment of core damage and the response of the containment during an accident.
- More accurate modeling of specific conditions of the plant site, including initiating events.
- Improved methodology for assembling the results and working backwards to specific risk contributors.

The more recent probabilistic safety studies in this country and Europe built on the foundation of the Reactor Safety Study, addressed its criticisms and incorporated these advances. The Indian Point study, in particular, represents the current state of the art.

UNDERSTANDING PROBABILISTIC RISK ASSESSMENT

Probabilistic risk assessment is considered the most advanced way to make practical decisions in a highly technical, complex society where risks cannot be eliminated, but must be controlled. It helps us understand which things are more likely than others to go wrong and provides a framework for deciding what, if anything, should be done about them.

People have sought ways to assess risk for centuries. Examples can be found in the insurance industries, the financial community, and others that deal with consumer protection. The challenge has been to assess risk systematically and to take into account unforeseen circumstances.

PRA allows you to do just that. Any accident can be examined, regardless of its likelihood. Uncertainties are clearly identified in the process. Analyzing "frequencies of occurrence" allows you to tag the most important accident scenarios among the many thousands examined. Human errors can be factored into the calculations as can complications unrelated to the accident itself that could change the levels of risk involved.

The use of PRA techniques is not limited to the nuclear industry. Others are already using a number of PRA techniques. That trend is expected to continue, with the nuclear industry in the vanguard.

Simply put, the PRA methodology asks three basic questions:

- What could go wrong?
- How likely is it that this will happen?
- If it happens, what are the consequences?

The answers to those questions help planners and decision-makers to determine what, if anything, should be done to reduce the likelihood that a particular type of accident could happen or to reduce the level of damage that could occur. The answers help isolate the factors that pose the most substantial chance of adversely affecting public health and welfare.

What Could Go Wrong?

In analyzing risk, we are attempting to understand the effects of taking or failing to take a certain course of action. Since an outcome of a course of action involves a whole sequence of events, the term "scenario" is generally used in place of "outcome."

Developing a "scenario" begins with identifying an event that could precipitate an accident. For example, lightning striking the roof of a building could start a fire which destroys the building. The lightning strike is called the "initiating event" or beginning of an accident sequence.

The next step in developing a scenario, after identifying an "initiating event," is to frame a series of questions that ask, "If such-and-such occurs, *what could happen next?*" Two kinds of answers are given in response to the question: one assumes that a safety barrier erected to prevent further damage from the initiating event works; the second answer assumes that the safety barrier fails.

Using our lightning strike analogy, if the lightning strikes the building, it could hit the lightning rod installed on the roof (the safety barrier works) or it could hit another part of the roof, such as an air conditioning unit (the safety barrier fails). If it hits the air conditioning unit, what could happen next? The regulator on the unit could shut itself down because of the power surge (the next safety barrier works) or the regulator could malfunction and cause a small fire to start in the wiring system (safety barrier fails). And so on. The questioning continues in this manner until every sequence of events that could result from the initiating event is identified.

The process of identifying all of the events in a particular sequence, and assuming that a safety barrier either works or fails, is called building an "event tree." In PRA, complete event trees are developed for all conceivable initiating events.

The next step in answering the basic question, "What could go wrong?" is to determine *how* each of the failures in the succession of safety barriers can happen. How, for example, did the fire start in the wiring system? This requires an examination of the subsystem or components that make up the safety barrier in order to identify those factors which could lead to failure of an entire barrier system. The results of investigating how the failures can happen are diagrammed on a "fault tree."

Fault trees are used to determine the likelihood of failure of the safety systems identified in the event tree. In developing the fault trees, consideration is given to component failure, maintenance action, human error, and other causes. Each system is examined in sufficient detail to determine the *frequency* of failures by looking at the reliability of each of the parts involved.

How Likely Is It That This Will Happen?

The likelihood of something going wrong is based on data about a particular element in the fault tree. The data may include, for example, operating records on equipment or systems.

If there is a large amount of such data, the likelihood of success or failure can be calculated with a high degree of certainty, or "confidence." When there is less data, the calculated frequency, or likelihood, is more uncertain.

These uncertainties are described in terms of "probabilities." In the context of this study a confidence level of 90% implies a probability that the parameter in question—frequency of occurrence of a particular event, for example—does not exceed a given value.

PRA studies include a rigorous mathematical assessment of uncertainty. The uncertainty of each element in the analysis is computed and is included in the final result. Thus, the likelihood of an event occurring is expressed in terms of frequency of occurrence *and* the level of confidence regarding the frequency. This format of communicating uncertainty is called the "probability of the frequency of occurrence."

PRA studies may consider literally millions of scenarios and their corresponding probability of the frequency of occurrence. These probabilities are tabulated individually, and the results are usually presented on a graph.

If only one confidence level or "probability level" is used, the results would form one curved line. It is more common to look at several confidence levels:

- The 1 in 10 chance, or 0.10 probability.
- The 5 in 10 chance, or 0.50 probability.
- The 9 in 10 chance or 0.90 probability.

These results are normally compiled on the same graph—three curved lines—and are called a "family of curves." The interval between the top curve and the bottom curve is the "uncertainty band."

To translate these confidence statements into everyday language, consider the frequency at the 0.50 probability level as a "best" estimate and the frequency at 0.90 probability as an "upper bound." Upper bound estimates mean that "it is almost certain that the frequency of occurrence will not exceed this value." For purposes of communication, therefore, discussions of risk assessment results will use statements such as "the *best* estimate of the frequency of this scenario is once every 10,000 years and the *upper bound* is once every 1,000 years."

If It Happens, What Are the Consequences?

The accident consequences of paramount concern are those affecting people's health. Consequences of accidents are evaluated using extensive computer programs to model accident scenarios, the conditions of the region around the site of the accident, population information, and any other relevant factors including protective measures that could offset some of the damage.

The results of the analysis of accident consequences are expressed as "damage levels"; e.g., numbers of injuries, numbers of fatalities. The consequence analysis is combined with the plant and containment analyses to arrive at the "probability of frequency of different levels of damage." The combined results of damage levels with frequency of occurrence are translated in graphic form into risk curves.

The risk curves from a PRA convey considerable information: for example, what is the frequency of accidents resulting in any immediate fatalities, or what is the frequency of accidents resulting in 100 or more injuries. The degree of confidence (probability) with which the result is stated—10%, 50%, or 90%—is also conveyed.

Frequencies of occurrence are not predictions. They are expressions of the collective knowledge and experience of the experts who performed the probabilistic risk assessment. Frequencies of occurrence suggest what the odds (probabilities) are of something actually happening and thus provide a basis for comparisons with other risks.

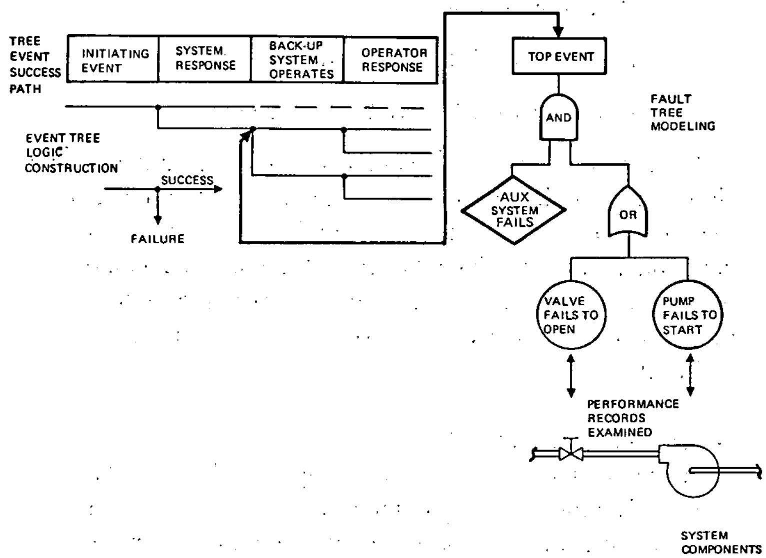
The risk curves link the likelihood that the accident could happen with the potential consequences of an accident. This is the proper way to view risk—probability coupled with consequences.

PROBABILISTIC RISK ASSESSMENT: SOME GRAPHIC ILLUSTRATIONS

The PRA methodology involves a rigorous process that organizes vast amounts of data through highly detailed analyses. The most concise way to display the key steps, the intermediate findings, and the final results is graphically. These are some very simplified examples of what is actually done in a fully developed study.

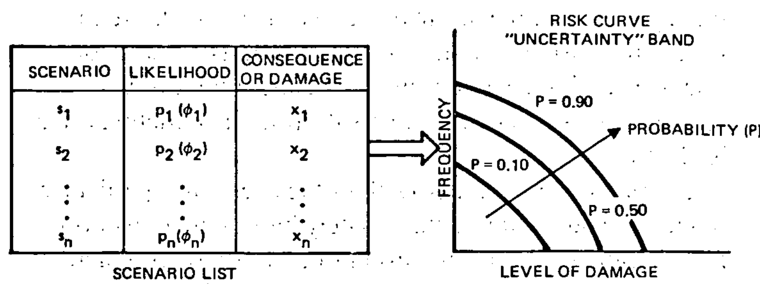
Quantifying Accident Sequence Probabilities

Event trees diagram what could happen as a result of an initiating event which could lead to an accident. Branches of the tree illustrate the success or failure of the series of safety barriers and frame the answer to the question "what could happen next?" When a failure is identified, it becomes the "top event" in a fault tree analysis. Fault trees trace the failure back to its root causes, primarily using reliability performance information, to determine how the failure occurred.



Risk Results

The list of accident scenarios, their likelihood of occurrence (including uncertainty), and their consequences number in the millions in a risk assessment study. After the confidence levels are determined, the computerized list format is translated into risk curves. Levels of damage (or consequences) are expressed in terms of numbers of people affected and the frequency of an accident occurring that could cause that kind of damage. Confidence levels for the results are labeled on the curves lines: 10% probability, 50% probability, and 90% probability. The interval between the 10% curve and the 90% curve is called the "uncertainty band" which shows the minimum and maximum potential consequences of an accident. The information in the scenarios list could be used to add more curves at other confidence levels if it were desired.



THE INDIAN POINT PROBABILISTIC SAFETY STUDY

A comprehensive safety study using PRA techniques was conducted for Indian Point Units 2 & 3, beginning in January 1980. A team of more than 50 experts was involved in the project, including nuclear engineers; systems analysts; probability theorists; mathematicians; risk analysts; computer specialists; experts in thermohydraulics, chemistry, radiological effects, meteorology, seismology and wind; and nuclear power plant operators and designers. This work was reviewed and discussed with an independent review board. The final study report, more than 6,000 pages long, was submitted to the U.S. Nuclear Regulatory Commission in early 1982.

The Indian Point study has two basic purposes:

- To provide a thorough assessment of public risk resulting from the operation of Indian Point.
- To identify the dominant contributors to that risk in terms of plant design and operation. In that connection, the study postulated a variety of equipment malfunctions including progressive failures of multiple engineered safeguards leading to melting of the reactor core and failure of the containment building.

The study team began by collecting extensive information from several sources:

- Specific Indian Point operating data, covering the plant and the site. Plant data included, for example, component performance records, maintenance duration reports, and initiating event analyses. Site data included comprehensive examinations of meteorology, terrain, and demographics.
- Operating data from other nuclear power plants. Numerous data sources were analyzed to establish a comprehensive data base. Sources of data included: (1) Licensee Event Reports and the NRC data summaries of these reports covering diesel generators, pumps, valves, and control rod drives; (2) the IEEE Guide to the Collection and Presentation of Electrical, Electronic, and Sensing Component Reliability Data for Nuclear Power Generating Stations; (3) the Reactor Safety Study; (4) the Nuclear Plant Reliability Data System; (5) the EPRI reports on Frequency of Anticipated Transients; and (6) the Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications.
- Expert judgment on equipment performance and accident initiators. Experts and their resources contributed additional insight and information about equipment performance and specific events that could initiate an accident or alter its course.

These data were examined using the PRA methodology discussed previously. In order to assess the possibility that public health could be endangered by an accident at Indian Point, the study identified accident sequences that could lead to release of harmful levels of radioactivity and isolated the dominant contributors to risk contained in those sequences.

The investigative process was exhaustive:

- Literally hundreds of thousands of accident scenarios were developed using the event tree/fault tree approach. Information about reactor operations and reliability of equipment was incorporated in the scenarios. These event trees and fault trees were used to evaluate various sequences leading to release of radioactivity.
- In order to isolate dominant risk contributors, initiating events from both internal and external causes were analyzed. Internal causes included plant malfunctions called "transients" and failures in heat removal systems such as loss of coolant accidents (LOCA's) which could lead to melting of the reactor core. External causes included earthquakes, fires, high velocity winds, tornadoes, floods, and aircraft accidents. Analyses of initiating events and accident sequences were not limited to hardware concerns; operator interaction with the plant systems and human response under accident conditions were also assessed.
- The response of the reactor core and containment under different core melt conditions was analyzed in extensive detail. The form and state of the damaged core and its interaction with structural materials, air, water, steam, and other reaction products were considered. The progress of core melt conditions was examined carefully; the pressure changes during a core melt and fluctuations in heat loads were quantified to define containment response; and release conditions for radioactive material were identified.
- The region around Indian Point was modeled, including information about population distribution and meteorological conditions. This information helped establish the level of risk which is partially dependent on wind speed and direction, the portions of the surrounding communities potentially affected, dispersion of radioactive material, and the like.
- After the dominant risk contributors were identified, along with their causes and probable frequency of occurrence, estimates were made of the potential damage to public health and safety. These estimates were compiled and displayed graphically as a family of risk curves which indicates the confidence level attached to the estimates.

Public Health Effects

The results of the Indian Point Probabilistic Safety Study focus ultimately on public health effects. The risks to public health discussed in this overview are early, or "acute," fatalities occurring within a short time after exposure, non-fatal radiation injuries due to exposure, thyroid cancers (most of which are treatable and non-fatal), and latent cancer fatalities occurring over a 30-year period. The results are highlighted here for three levels—any effect, 100 effects, and 1,000 effects. In like manner, frequency of occurrence results have been computed for other levels of effects.

The results are also presented as "best" and "upper bound" estimates. By presenting two estimates, this analysis provides a more comprehensive picture of risk than would be the case if only a single health effect, a single level of effect, or a single confidence level were depicted.

Indian Point Unit 2. The most significant health effect, in terms of near-term impact, is acute fatalities. The estimates vary depending on the level of consequences analyzed. The best estimate for any effect is once in 17 million (1.7 million)* years of reactor operation. The best estimates of frequencies of occurrence for 100 and 1,000 effects are once in 100 million (4.8 million) years and once in a billion (29 million) years, respectively.

For radiation injuries, the best estimate for any effect is once in 370,000 (59,000) years. The best estimates for 100 effects are once in 2.9 million (290,000) years, and for 1000 effects once in 110 million (2.9 million) years.

In addition to the immediate health effects of an accident, delayed effects were also examined. For example, the best estimate of any thyroid cancers occuring is once in 2,500 (1,000) years of reactor operation. The best estimates for 100 effects are once in 5,900 (1,400) years and for 1,000 effects once in 12,000 (2,700) years.

Finally, the question of latent cancer fatalities occurring over a 30-year period was investigated. The best estimate for any effect is once in 3,000 (1,000) years. The best estimates for 100 effects are once in 5,000 (1,400) years, and for 1,000 effects once in 10,000 (2,400) years.

Indian Point Unit 3. The results for Indian Point 3 differ somewhat from Unit 2 due to some differences in design and equipment. The best estimate for any acute fatality is once in 83 million (10 million) years. The best estimates for 100 effects are once in 310 million (45 million) years, and once in 6.3 billion (290 million) years for 1,000 effects.

For radiation injuries, the best estimate for any effect is once in 2.6 million (330,000) years. For 100 effects, the best estimate is one in 20 million (2.4 million) years, and for 1,000 effects, the best estimate is once in 310 million (28 million) years.

*The numbers in parentheses are the upper bound estimates.

The best estimate for any latent thyroid cancers occurring is once in 12,000 (3,100) years. The best estimates for 100 effects are once in 63,000 (7,700) years, and for 1,000 effects once in 100,000 (12,000) years.

The best estimate for any latent cancer fatalities occurring over a 30-year period is once in 20,000 (5,000) years. The best estimates for 100 effects are once in 55,000 (8,000) years, and for 1,000 effects once in 100,000 (12,000) years.

The likelihood that an accident would cause any public health consequences is remote. Upper bound estimates indicate that an accident causing any acute fatality is once in 1.7 million years and that an accident resulting in 100 or more latent cancer fatalities is once in 1400 years. Information on accidental fatalities and latent cancer fatalities from *non-nuclear* causes provides some perspective on these potential health effects. For example, every year, based on the national average, there will be at least 100 accidental fatalities within a 10-mile radius of Indian Point and at least 30,000 cancer fatalities within a 50-mile radius of the plant, all unrelated to nuclear power.

The Results in Perspective

The risk assessment for Indian Point identified the elements of an accident that would need to be present for any fatalities to result.

- *The Reactor Core Must Melt.* Coolant must be maintained in the reactor core to avoid fuel damage. Therefore, nuclear plants contain several backup cooling systems that can be called on to cool the core if the primary system should stop functioning. Only if these "emergency core cooling systems" should fail would some of the fuel rods melt or be damaged, causing a release of fission products into the reactor vessel and reactor coolant system.

  If there should be a core melt, this by no means suggests that there will necessarily be a significant release of radioactivity to the environment outside the plant. The containment structure of a nuclear plant is designed to contain the radioactive material and prevent such releases. A core melt by itself constitutes no real threat to public health and the study indicates that most core melts would be contained without significant release of radioactive material to the environment.

- *The Containment Must Fail or Be Bypassed.* If the containment serves its function, a core melt would only lead to some leakage of radiation around some of the piping passageways that connect the containment with other buildings in the plant. The risk of those small leaks would be of little or no consequence to the public health.

If the containment were to fail, the amount and type of risk to public health would depend upon conditions such as wind speed and direction; how quickly the radioactive "plume" is dispersed; and the effectiveness of protective steps like sheltering or evacuation. The possibility of bypassing the containment was also examined. Out of all accident categories studied for Indian Point, only about 1 in 1,000 core melt accidents leads to a release which could potentially cause any early fatalities.

A study with the scope and level of detail of the Indian Point safety assessment produces an extremely large body of information and results. Thousands of accident scenarios have been identified and their likelihood and consequences discussed. This overview presents only the key findings about public health effects and the safety of Indian Point.

The application of risk assessment to nuclear safety analyses, while adding immeasurably to our understanding of nuclear safety, has perhaps colored our perception of what the risks truly entail. Because comparable assessments have yet to be performed to the same level of detail for other energy sources and other industries, nuclear safety is being weighed in a vacuum.

Risk analyses in one sense may counteract their intended purpose. Events which are beyond reasonable belief tend to assume a degree of reality when they are analyzed in minute detail. Detailed analyses make people aware of possible hazards which in all probability will never result in injury. Ironically, risk analyses may demonstrate that certain risks are exceedingly remote yet fear of those same risks may increase by that very demonstration. The initial perception of "rare and remote" evolves into a growing uneasiness that "something just may happen." On the other hand, far greater risks unrelated to nuclear power may not be viewed with concern simply because they have not been so intensively investigated. Nuclear power—its safety and its risks—should be considered in that context.

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