UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	Docket Nos. 50-247-SP
CONSOLIDATED EDISON	)	50-286-SP
OF NEW YORK (Indian Point, Unit 2)	)	
	)	
POWER AUTHORITY OF THE STATE	)	
OF NEW YORK (Indian Point, Unit 3)	)	

DIRECT TESTIMONY OF ROGER M. BLOND AND FRANK H. ROWSOME
SUMMARY RESPONSE TO COMMISSION QUESTION POSED TO THE BOARD

IV.A. CONCLUSION

Summary Response to Commission Questions Posed to the Board:

What risk is posed by severe reactor accidents at IP 2/3 as they are currently designed and operated?

Q.1 Please state your name and your position with the NRC.

A.1 My name is Roger M. Blond. I am Section Leader for the Accident Risk Section of the Reactor Risk Branch of the Division of Risk Analysis of the Office of Research.

Q.2 What are your responsibilities in that position?

A.2 I am responsible for providing technical and managerial direction in developing methods and research in accident risk analysis and in performing applications in risk assessment.

Q.3 Have you prepared the statement of your professional qualifications?

A.3 Yes, the statement of my professional qualifications is attached to this testimony.

Q.4 Please state your name and your position with the NRC.

A.4 My name is Frank H. Rowsome. I am Deputy Director of the Division of Risk Analysis in the Office of Nuclear Reactor Research.

Q.5 What are your responsibilities in that position?

A.5 I assist the Director in planning and managing the research group in risk assessment, probabilistic safety analysis, operations, research, reliability engineering, and related regulatory standards development.

Q.6 Have you prepared a statement of your professional qualifications?

A.6 Yes, the statement of my professional qualifications is attached to this testimony.

Q.7 What is the purpose of your testimony?

A.7 The purpose of my testimony is to summarize the staff's testimony on the risk at Indian Point to respond to the Commission questions. As was discussed in the introductory material, reactor accident risks can be described in different ways. Risk can be represented by one number which is calculated as a simple summation of the accident probabilities times the associated consequences; this one number is generally referred to as the expected risk. Risk can also be represented by a curve relating probability to exceeding a given level of consequence; this curve is known as the Complementary Cumulative Distribution Function (CCDF). Appendix A of NUREG-0715 describes how such figures are constructed. In addition, risk can be represented by the level of uncertainty associated with the probability and consequence estimates; as was done in the Indian Point Probabilistic Safety Study. Common to all of these definitions of risk is the concept of accident probability and consequence. S. Israel, R. Budnitz, and B. Buchbinder in Section III.A described accident probabilities; J. Meyer in Section III.B described the approaches and assumptions used to generate the magnitude and characteristics of the radioactive release to the atmosphere; and S. Acharya in Section III.C described the calculations associated with the consequence estimates. Building upon each section, the probabilities and consequences are generated for seven different risk measures. These are early fatalities (occurring within a year after exposure); early injury; delayed or latent cancer fatality; thyroid cancer; genetic effects; offsite property damage costs; and land contamination. In addition, perspectives are given on the individual as well as the societal risks.

Probabilistic risk assessment attempts to provide the relationship between the probabilities of the accidents and the consequences. To presume a release occurs without accounting for its likelihood is a misrepresentation of risk. It is possible to contrive scenarios by which most of the radioactive material could be released to the atmosphere and would be transported to the most populous location where it could do the most harm. However, the chance of this realistically occurring has to be factored into the analysis. This realistic analysis is the objective of probabilistic risk assessment.

Q.8 You defined risk as probabilities and consequences, what is included in the probability portion of risk?

A.8 The probability portion of risk includes the probabilities associated with the accident sequence occurrence, and the probabilities associated with the magnitude of the radioactive material released to the environment, and the probabilities associated with the magnitude of the consequences.

Q.6 What is included in the probabilities associated with the accident occurrence?

A.6 There are several factors which go into making up estimates of the probabilities of potential accidents. These factors are given in the following formula:

$$P_{\text{accident}} = P_{\text{initiator}} \times P_{\text{system failures}} \times P_{\text{containment failure}}$$

where P_{accident} is the probability of the accident, $P_{\text{initiator}}$ is the probability of the initiating event or damage state, $P_{\text{system failures}}$ is the probability of a sequence of systems failures, and $P_{\text{containment failure}}$ is the probability of the containment failure.

This then defines a sequence of events in terms of probabilities--starting from the initial accident cause and working through the systems and containment response. There is thus a dependence in moving from one factor to the next.

For example, one of the more probable severe accident sequence at either Indian Point unit prior to the recent fixes was caused by a fire which fails a pump seal causing a small-break loss-of-coolant accident. The fire simultaneously affects other safety and containment systems. Thus, the fire prevents the emergency core cooling systems from operating and this leads to core melt. The fire also disables the containment heat removal systems. In terms of the above equation, $P_{initiator}$ is the fire, considered an external common cause failure, causing a small-break loss-of-coolant accident. The probability of such an event was assessed by the IPPSS to be about 2×10^{-4} per reactor year for Unit 2. $P_{system\ failure}$ and $P_{containment\ failure}$ are both assumed to be 1.0 given that the fire occurs. Therefore, the probability associated with this accident occurrence is $2 \times 10^{-4} \times 1.0 \times 1.0$ which equals two chances in ten thousand per reactor year for Unit 2.

Q.7 What is included in the probabilities associated with the magnitude of the radioactive material released to the environment?

A.7 The probabilities associated with the magnitudes of the radioactive material released to the environment are currently assumed to be synonymous with the probabilities associated with the specific containment failure mechanisms. If the containment fails by overpressurization--large amounts of radioactive materials can be released; however, if the containment fails by basemat melt-through--relatively little radioactive material would be released via the atmosphere.

Moving along with the above example, we have already assumed containment failure with a probability of 1.0. However, even assuming no containment heat removal, according to the Section III.B there is about a 60% chance that the containment fails by basemat melt-through, and about a 40% chance that the containment fails by late overpressurization. Containment overpressurization is necessary to have the potential for a large release. Thus the probability associated with this accident having a large release of radioactive material directly to the atmosphere is $2 \times 10^{-4} \times 0.4$ which equals 8×10^{-5}.

Q.8 What is included in the probability associated with the magnitude of the consequences?

A.8 The probability associated with the magnitudes of the consequences includes the probabilities associated with the following kinds of factors: weather conditions, numbers of people exposed (wind direction), emergency response, and health effects given exposure. If there is a 10 percent chance that the wind blows the radioactive material in a direction where there are no people given the accident, then the absolute probability associated with no health effect consequences from the above example is $8 \times 10^{-5} \times 0.1$ which equals 8×10^{-6} per reactor year. This combination of probability and consequence (8×10^{-6} and 0) represents one potential point of the risk at Indian Point.

Q.9 For the Indian Point plants, what is the relationship between the accident probabilities and the potential release of radioactive material?

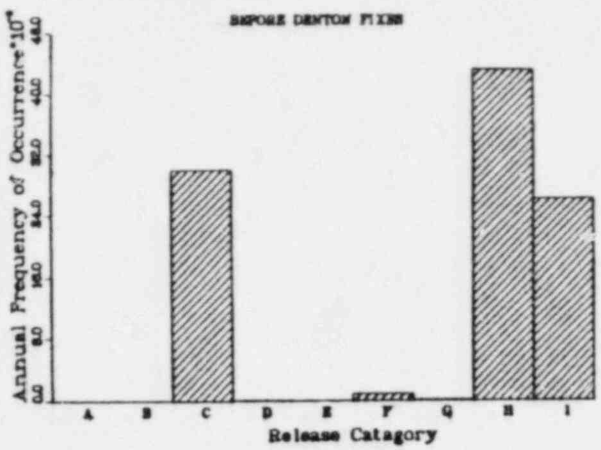
A.9 As has been explained in previous testimony, there is a spectrum of accident releases postulated for the Indian Point reactors. This spectrum ranges from accidents like the Release Category I event (given in Table III.C.3) which would release very small fractions of the radioactive material, to accidents like the Release Category A event which would release very large fractions of radioactive material. For the Release Category I event, there are little or no public health and safety consequences, whereas for the A release the consequences could be very severe.

Based upon the Indian Point Probabilistic Safety Study (with Sandia National Labs and staff refinements as described in III.A), a set of probabilities were derived for each of the damage states. These probabilities, shown in Table III.B.1 under the "Before Fix" columns, represent the plants as analyzed circa 1981. As explained in the testimony of Messrs. Bookbinder Budnitz, and Rowsome Unit 2 has now been modified to reduce the seismic fragility, reduce the fire vulnerability and develop procedures to shutdown for severe hurricanes; in addition, Unit 3 has been modified to reduce fire vulnerability. To reflect these modifications, new probabilities of the damage states are shown in Table III.B.1 under the "After Fix" Columns.

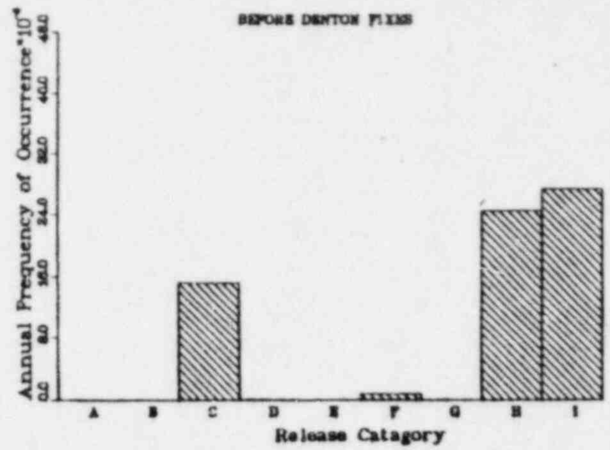
Figure IV.A.1 summarizes the analyses for both before and after fix and shows the three dominant accident sequence/containment failure categories. For either unit the most probable accidents, given that a core melt occurs, is either the case in which the containment holds, Release Category I, or there is an eventual release through the containment basemat, Release Category H. For both of these accidents there is minimal offsite health and safety impact. There would not be any early health effects and contamination levels would be so low that emergency response would probably not be mandated under current Environmental Protection Agency protective action guideline dose projection recommendations. The only significant accident probability scenario with a large release/consequence potential is the long-term overpressurization category, C. Even though there is the potential for large releases, there would be significant warning times (at least 8 hours) for the public to take protective measures. Large release scenarios which will occur quickly such as categories A and B have only about one chance in a thousand of occurring given a core melt occurs. Thus, the risk importance of these scenarios is significantly reduced.

Comparing the "Before Fix" to "After Fix" charts reveals that the major change due to the fix was in reducing the probability of the C damage state. This is due to the emphasis on external events which dominate the risk.

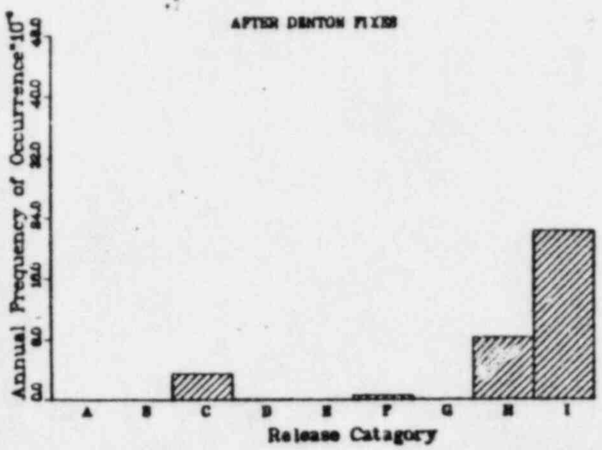
Containment Frequency of Release For Unit 2



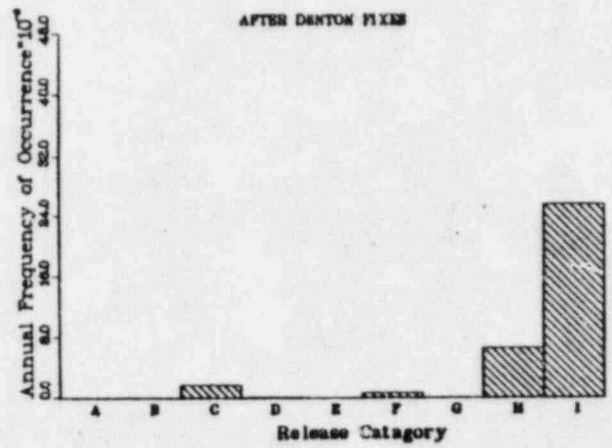
Containment Frequency of Release For Unit 3



Containment Frequency of Release For Unit 2



Containment Frequency of Release For Unit 3



Release Category	Failure Mode
A	Large seismic event β* (containment collapse)
B	Event V and all α (alpha) failure modes
C	All long-term δ (delta) overpressurizations and SGTR event
D	All early γ (gamma) hydrogen burns (no sprays)
E	All late γ (gamma) hydrogen burns (no sprays)
F	All early γ (gamma) hydrogen burns (with sprays)
G	All β (beta) failure modes (failure to isolate containment)
H	All ε (epsilon) basemat penetration modes
I	All conditions for which containment failure does not occur.

Figure IV.A.1 Release Category Contribution To Probability of Core Melt

Q.10 How do the accident probabilities and release magnitudes relate to the risk at Indian Point?

A.10 A perspective on the accident initiators will provide insights into the initial conditions which will be expected in the reactor systems.

More than 90 percent of the risk comes from small-break loss-of-coolant accidents. Transients and large-break loss-of-coolant accidents have very little risk significance at Indian Point. It should be noted, however, that transients would cause small breaks in the pump seals. These small breaks are a dominant contributor to risk. These transient-induced loss-of-coolant accidents are counted as small break events. To gain an appreciation for the relationship of intend/external events, containment heat removal and containment failure to the Indian Point plants, a set of bar chart figures are given below. An appreciation for the importance associated with internally versus externally caused accidents is shown in Figure IV.A.2.a. For Indian Point, seismic events, fires and hurricanes are very important to the risk perspectives they pose far more risk than the internal initiators.

The second factor in the accident probability formula is the probability of system failure. This is a very important term in the equation for it can give insight into potential system importances which can influence accident prevention. System interactions are also investigated at this stage of the analysis. In addition, human failures as well as machine hardware failures are considered. In the risk model, the human error contribution is treated as an integral part of the failure analysis and cannot easily be broken into its component parts.

The third factor in the accident probability formula is the containment failure probability. Figure IV.A.2.b displays the contribution of containment heat removal to the risk. For the Indian Point containments, both the containment sprays and the fan coolers must fail in order to fail the containment directly to the atmosphere. As such, if either system is operable, the risk is very small. The last column on the chart, Figure

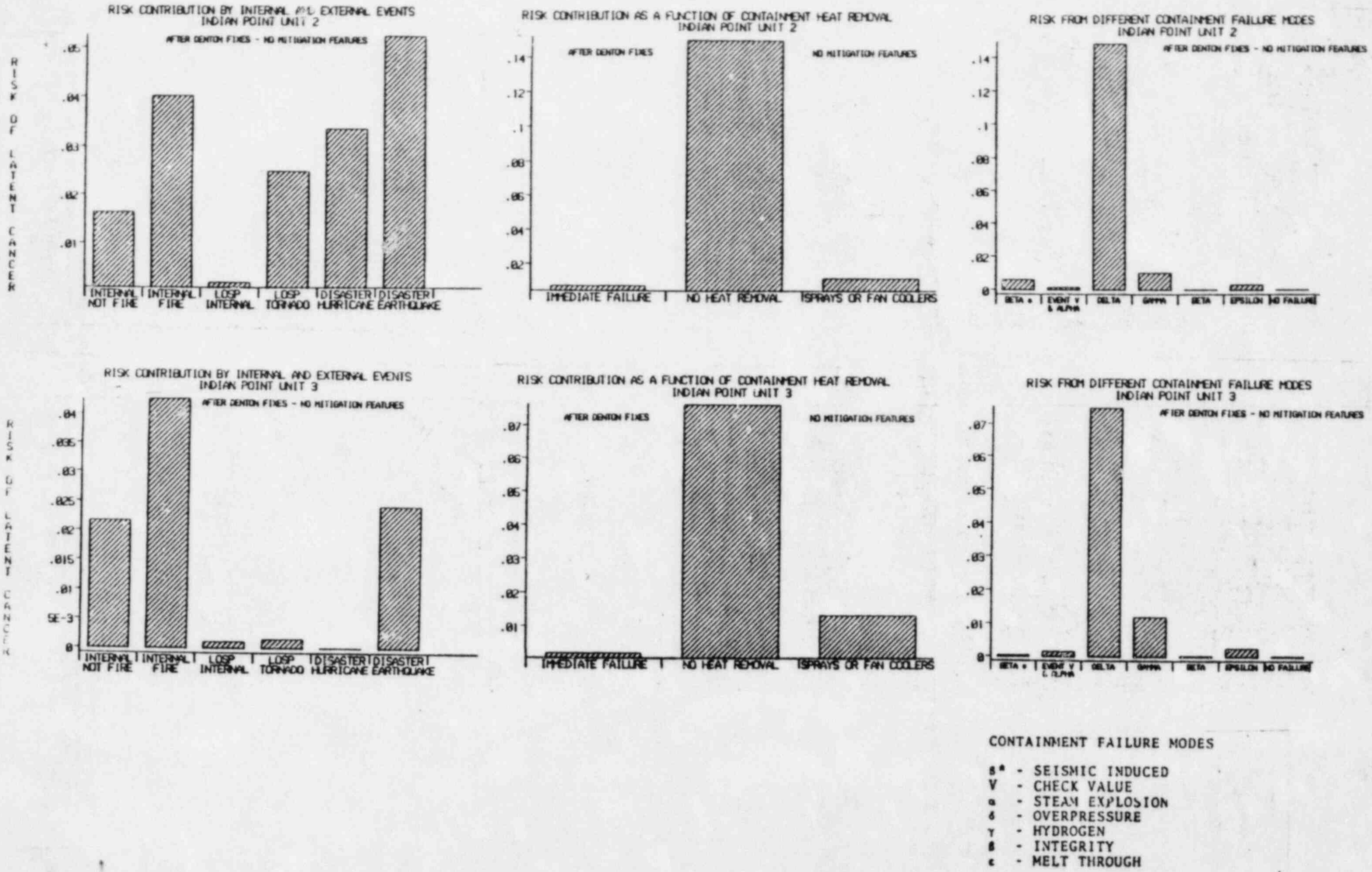


Figure IV.A.2 Relationship of Indian Point Units 2 and 3 to Accident Risk (Latent Cancer Fatalities)

IV.A.2.c, shows the contribution of containment failure mode to risk. The dominant risk containment failure mode is by containment overpressurization.

The following perspective on the Indian Point designs should now be apparent. Risk is dominated by small LOCAs, from external common cause events, where no containment heat removal is possible, and the containment fails by overpressurization. This picture is representative of the Indian Point power plants as analyzed in the IPPSS. Other designs would probably have significantly different dominant accident characteristics. With this description of the plants vulnerabilities, the question of the risk at Indian Point can now be addressed.

Q.11 What is the expected risk at Indian Point?

A.11 Section III.C presented a comprehensive evaluation of the risk at Indian Point. Ten risk measures are presented. Eight are relevant to health impacts and two are related to economic impacts of the accidents. Three emergency response scenarios have been analyzed which represent the expected response under various conditions and strategies. As is explained in section IV.B, the evacuation-relocation strategy is believed to be the most representative for those accidents which are not caused by regional disasters. When accidents are caused by regional disasters the late relocation strategy is believed to be the appropriate model.

It is interesting to note that if the early relocation strategy were used as the base case instead of the evacuation-relocation strategy, the results would be just about the same. Section IV.B explains why this is so. It is important to recognize that the details associated with the evacuation versus a relocation strategy will have little to do with the risk at Indian Point. Upon examination of the probabilities given in table III.C.4 and the conditional consequences given in table III.C.5 it is concluded that the risk at Indian Point is dominated by release category C caused by earthquakes and hurricanes.

Tables III.C.6 and III.C.7 give the expected risk for both Indian Point units 2 and 3 respectively.

Some general comments can be made concerning the expected risks associated with reactor accidents at Indian Point. First, the expected risk for all of the measures associated with health effects (i.e., early fatalities, early illnesses, latent cancer fatality, thyroid cancer, and genetic effects) is about one per reactor year. Second, the expected risk associated with offsite property damage is on the order of a million dollars per year from reactor accidents at Indian Point. These numbers are indicative of the risks associated with Indian Point. They include both the probability of the accidents and the associated consequences. Therefore, they represent the annual threat that the plants pose to the public.

Q.12 Explain the importance of the risk curves (complementary cumulative distribution functions) for Indian Point.

A.12 The probability/consequence relationship is very important to the concept of risk. The following sections summarize the risk curve results for the measures of risk that have been previously calculated. As discussed in the introduction, there are large uncertainties associated with the results presented. The numbers given are best estimate values and should be recognized as such.

Early Fatalities

For the most severe accidents, there is a very small probability that sufficient radiation could be released to cause potentially lethal exposures. For comparison purposes, supportive treatment will be assumed in this summary. Lethal exposures would probably be limited to a couple of miles from the reactor. However, for adverse meteorological conditions (e.g., downwind precipitation), it is possible to have very localized high concentration areas that could give lethal exposures at distances out to about 30 miles from the reactor.

As can be seen from Figures III.C.1 and III.C.12, there is a probability of about 5×10^{-6} and 1×10^{-6} associated with consequences of at least 1 early fatality caused by potential reactor accidents at Indian Point Units 2 and 3, respectively. This corresponds to the probability associated with accident sequences having large releases multiplied by the probability that the wind was blowing in a populated area while transporting sufficient radioactivity to cause lethal exposures.

Figures III.C.1 and III.C.12 also display the importance of the emergency response strategies to early fatality risk. See also Section IV.B for a discussion of the effectiveness of emergency response in terms of risk.

In addition to the emergency response considerations, another important aspect of the early fatality risk concerns the ability to provide a sufficient number of people with supportive medical treatment. Analysis of the Section III.C curves indicate that the risk of early fatality is about a factor of three higher if supportive medical treatment is not administered.

The probability associated with at least 100 fatalities in Figure III.C.1 is approximately the same as the probability associated with one fatality. This is because there are a number of weather conditions and directions (populations exposed) which will cause up to about 100 lethal exposures. The "largest" number of fatalities calculated at Indian Point Units 2 or 3 is about 30,000 to 40,000 people and has a probability associated with it of about one chance in one hundred million and one chance in a billion per reactor year for Units 2 and 3, respectively. This "largest" number of fatalities is associated with the simultaneous occurrence of the "largest" release categories, the "worst" weather conditions, the wind blowing in the direction of the "highest" population, and a very pessimistic emergency response assumption. The adjectives "largest," "worst," and "highest" are put in quotation marks to indicate that they represent the maximum values generated in the calculation. This is not to say that larger numbers could not be generated. Indeed it is conceivable to devise conditions which will increase the "maximum" somewhat, however the probabilities associated with such increases should also diminish.

Individual risk of early fatality as a function of distance is shown in Figure III.C.23. This figure gives some perspective on the risk imposed by Indian Point Units 2 and 3 as a function of distance. The important message conveyed in these figures is the very low (10^{-6} to 10^{-11}) absolute probabilities that have been projected for an individual's risk of being exposed to a lethal level of radiation.

Early Injury

The risk measure of early injury is defined to be those injuries which would have visible symptoms and would probably warrant some medical attention shortly after the accident. Types of injuries considered are prodromal vomiting, diarrhea, and respiratory impairment. Calculations indicate that for the largest releases there would be about a factor of ten more early injuries than early fatalities. In addition, even for large releases, it would be expected that early injuries would be limited to within about 10 miles of the reactor. However, for the adverse meteorological conditions, early injuries could occur out to about 50 miles. The complementary cumulative distributions functions for early injury are shown in Figures III.C.4 and III.C.15. Individual risk curves versus distance are given in Figure III.C.25.

Latent Cancer Fatality

Radiation exposures can increase the number of cancer fatalities in the exposed population. The model used to predict the increase in cancer incidence is the Reactor Safety Study modal which is similar to the linear-quadratic model recently recommended by the Biological Effectiveness of Ionizing Radiation (BEIR) III Committee of the National Academy of Sciences.

Figures III.C.6 and III.C.17 present the probability/consequence curves for latent cancer fatalities. The unique feature concerning the latent cancer fatalities is that the cancer fatalities would begin after some latency period, occur over a period of many years, and would include many different types of cancer. As such, it is not appropriate to simply add the latent cancer fatality risk to the early fatality risk which occur within the first year without accounting in some manner for this fundamental difference. However, there would be expected to be about 10 times more latent cancer fatalities than early fatalities. One technique to compare early fatalities to latent cancer fatalities which has been used in the past (i.e., WASH-1400) is to divide the total latent cancers by 30 years, which is the period during which most of the cancers will be occurring, thus generating a yearly cancer fatality rate due to the reactor accident. As such there would probably not be a statistically significant increase in the cancer fatality rates of the population at risk.

The risk to an individual of latent cancer fatality is presented in Figure III.C.26. This figure shows that there is about a two order of magnitude drop in risk from about one mile to about 50 miles from the reactor. In addition, as can be seen in the figure, emergency response assumptions (evacuation/relocation) have a minimal impact on the total latent cancer fatality risks.

Thyroid Nodules

Based upon current (WASH-1400) source term assumptions, large amounts of radioactive iodine will be released in the most severe reactor accidents. If the radioactive iodine is deposited internally in the human body, it will be concentrated preferentially in the thyroid gland. As a result, the radiation dose to the thyroid is likely to exceed the dose to the rest of the body, and thyroid damage is likely to adversely affect more individuals than any other accident-induced radiation effect. The model used in this study predicts approximately 334 thyroid nodules per million person-rem, based upon thyroid dose. Of the predicted nodules, about one-third are estimated to be cancerous. Both benign and malignant nodules can be medically treated with good success.

It has been conservatively assumed that one-tenth of the cancerous nodules would be lethal. Figures III.C.8 and III.C.19 present the probability and consequence estimates for thyroid cancer fatalities for the Indian Point reactors.

The number of thyroid nodules predicted in the largest accidents would be approximately equal to the normal annual incidence rate of thyroid nodules in the exposed population. Therefore, the largest accident would approximately double the normal incidence. This effect would probably be detectable in the population at risk.

Genetic Effects

It is believed that radiation exposure would increase the mutation rate of genetic disorders. Radiation-induced mutation would not be any different than mutations that occur in nature. The effects would include such obvious effects as albinism or they can be almost undetectable. The model used in this study assumes 260 cases of genetic abnormalities per million person-rem, based upon whole-body dose. Figures III.C.9 and III.C.20 present the Indian Point total whole-body person-rem. It is estimated that there would be an increase of about 2 percent of the background genetic abnormality rate for the most severe accidents, for the population at risk.

Property Damage

All of the preceding risk measures are related to human health impacts of potential reactor accidents. There are actions which could be taken to reduce the risk of health effects of radiological exposure. However, the actions have costs associated with them. There are five actions which could be taken to reduce the health risks. The models used in this analysis assign costs to each of these actions based upon Indian Point specific data: first, there is a cost associated with people evacuating; second, there is a cost associated with the temporary relocation of people from those contaminated areas within which projected doses for continued exposures are above acceptable levels; third, there is a cost associated with the loss of benefits from property that must be interdicted for long periods because it cannot be reasonably decontaminated; fourth, there is a cost associated with decontaminating an area; and fifth, there is a cost associated with disposing of contaminated agricultural products (i.e., dairy and crops).

To estimate the costs, criteria are needed for setting acceptable dose levels. If criteria are set at very low dose levels, the health impacts would be reduced, but the costs associated with the five actions would be very high. Conversely, if the acceptable dose levels are set very high, then the costs associated with the actions would be minimal, but the health impacts would increase. An arbitrary acceptable dose level for relocation was chosen in the Reactor Safety Study to be 25 rem in 30 years. This value represents a tradeoff between health effects and costs. If the value were increased to 50 rem in 30 years, the expected costs would be decreased by a factor of 4; and the expected latent cancer fatalities and genetic effects would increase by about 10 percent. If the value was decreased to 10 rem in 30 years, the expected costs would increase by a factor of 2.5, and the latent cancer fatalities and genetic effects would decrease by about 10 percent. The predicted costs associated with reactor accidents at Indian Point are shown in Figures III.C.10 and III.C.21. The major contribution to the cost from the largest accidents is from interdicting those areas where reasonable decontamination procedures could not reduce the contamination to acceptable levels. Costs associated with decontamination and relocation expenses would also be significant. The other costs, associated with agricultural losses and evacuation, would not be significant in comparison to the total costs.

Land Contamination

One of the most important contributions to the property damage costs is from the land area that would require interdiction. A very small land area (less than one square mile) would probably need to be interdicted for the projected accidents. Figures III.C.11 and III.C.22 give the complementary cumulative distribution functions for the land contamination risks. As with the property damage estimated, the size of the land areas requiring protective measures will be correlated to the criteria which is used to assess the damage.

The health effects associated with liquid pathways from direct deposition and from groundwater sources have also been assessed in Section III of this testimony. It was concluded in Dr. Codell's testimony that there would not be significant health effects from liquid pathways in comparison to the above health effects.

Q.14 Does this conclude your testimony?

A.14 Yes.

PROFESSIONAL QUALIFICATIONS
ROGER M. BLOND
U.S. Nuclear Regulatory Commission

I am Roger M. Blond, Section Leader of the Accident Risk Section, Reactor Risk Branch, Division of Risk Analysis, Office of Research. I have been with the NRC since August 1974. In my present position, I am responsible for providing technical and managerial direction in developing methods and research in accident risk analysis and in performing applications in probabilistic risk assessment. This work includes: (1) developing risk models for calculating the physical processes and consequences of reactor accidents; (2) rebaselining accident consequences and reactor risk; and (3) developing value/impact analysis methods for reactor design improvements.

In addition to the Section Leader position, I have the following responsibilities:

- I am the Chairman of the International Benchmark Exercise on Consequence Modeling, sponsored by the Committee on the Safety of Nuclear Installations, of the Nuclear Energy Agency, Organization of Economic Cooperation and Development. As Chairman, I am responsible for organizing and directing the comparison study which includes the participation of 30 organizations representing 16 countries. The study was chartered to compare the large number of computer models that had been developed to calculate the offsite consequences of potential accidents at nuclear power facilities.

- I am responsible for developing the technical rationale for the development of improved siting criteria. This work includes the development of a set of representative potential reactor accident source terms, and a full parametric study of all the factors important to siting considerations from the risk perspective:

- I am a member of the Technical Writing Group of the IEEE/ANS PRA Procedures Guide - NUREG/CR-2300. This effort is developing a source document on PRA techniques. I am a co-author of the consequence modeling sections of the report.

- I am a member of the Department of Energy Working Group on Probabilistic Risk Assessment.

- I am a member of the NRC Incidence Response Center's Emergency Response Team.

In addition, I am directly involved in the development of a technical rationale for the NRC's Safety Goal, emergency planning and response, and numerous issues and questions which continuously arise in risk assessment.

I am also a lecturer on consequence modeling and accident analysis for the NRC Training Course on Probabilistic Safety and Reliability Analysis Techniques, for the IAEA Training Course on Nuclear Power, and for the George Washington University Seminar on Probabilistic Risk Assessment.

Risk Analyst

Before being selected for the Section Leader position, I was Senior Risk Analyst in the Office of Research. I was responsible for the following areas:

1. Consequence modeling research and development;
2. Performing and reviewing probabilistic risk assessments;
3. Siting and emergency planning and response criteria development; and
4. Integrating probabilistic risk assessment techniques into the regulatory and licensing process.

1. Consequence Modeling Research and Development

I was responsible for revising the consequence model that was developed for the Draft Reactor Safety Study. During the course of that effort, I developed the following modeling approaches and techniques which were used for the final Reactor Safety Study consequence model (CRAC) and are documented in Appendix VI of WASH-1400 and the CRAC User's Guide:

1. Meteorological sampling technique;
2. Diffusion modeling technique;
3. Time-varying meteorological model;
4. Depletion approach;
5. Finite cloud correction model for gamma shine;
6. Economic model;
7. Statistical sampling technique;
8. Emergency response model;
9. Property damage model; and
10. Population treatment.

After the completion of the Reactor Safety Study, I developed the following modeling techniques which have been incorporated into the CRAC-2 computer code and documented in the CRAC-2 User's Guide:

1. Revised comprehensive emergency response model;
2. Importance sampling for meteorological data and terrain diffusion model;
3. Revised dosimetry and health effects review; and
4. Comprehensive results display package.

I also performed numerous sensitivity and parametric studies on the models and input used in the consequence model and was responsible for an extensive research program to investigate the significance of various related phenomena to risk. This research involved from five to ten contractor personnel. I also have been responsible for preparing and defending the research program and budget in consequence modeling and emergency planning before the Senior Contract Review Board and the Advisory Committee for Reactor Safeguards.

2. <u>Performing and Reviewing Probabilistic Risk Assessments</u>

I was responsible for all of the risk calculations performed for the final Reactor Safety Study. At the completion of the study, I responded to critiques and questions concerning Probabilistic Risk Assessment from within the NRC, Congress, other Federal agencies, contractors and vendors, intervenors, state and local governments, utilities, and foreign governments. I have also performed risk studies or comparisons for the following analyses:

1. Task Force Report on Interim Operation of Indian Point;
2. Indian Point and Zion Site Risk and Alternative Containment Concepts Study;
3. Hatch consequence study;
4. Three Mile Island Potential Accident Consequence Study and Source Term Study;
5. Generic Environmental Statement on Mixed Oxide consequence study;
6. Anticipated transients without SCRAM consequence study;
7. Diablo Canyon Risk Assessment review; and
8. Clinch River Breeder Reactor consequence analysis review.

I have been responsible for advising and reviewing the following foreign risk assessments:

1. Norwegian Energy Study
2. Swedish Reactor Safety Study
3. German Reactor Safety Study
4. British Windscale and PWR Inquiries

In addition, the Norwegian Government personally invited me to Norway to review the approach and assumptions used in their study.

3. Siting and Emergency Planning and Response Criteria Development

I was the research consultant and member of the NRC/EPA Task Force on Emergency Planning. For the work of the Task Force, I was responsible for formulating the rationale for the emergency planning basis criteria and was the principal author of the Task Force Report on Emergency Planning (NUREG-0396). I also was responsible for developing the Emergency Action Level Guidance (NUREG-0654, Appendix 1) which establishes consistent criteria for declaring emergencies based upon plant parameters.

I performed a study on the cost/benefit of issuing Potassium-Iodide to the general public. Based on this report (NUREG/CR-1433), Potassium-Iodide is not being stockpiled for public distribution. In addition, I have performed numerous studies on emergency protective measures such as sheltering versus evacuation. I also developed the Three Mile Island Emergency Contingency Plan at the time of the accident.

I developed a ranking of high population sites which has been used to designate potentially high risk contributors.

4. <u>Integrating Probabilistic Risk Assessment Into the Regulatory Process</u>

I have provided technical direction on consequence modeling to the regulatory and licensing process for the following areas: Perryman Alternative Site Review; Environmental Impact Statement for Class 9 Accidents; Liquid Pathway Generic Study; in understanding the course and importance of potential accidents; and in source term development. I have on numerous occasions presented the results of my work on consequence modeling and emergency planning and response to other Offices within the agency, other organizations, the Advisory Committee on Reactor Safeguards, and the NRC Commissioners.

Science Applications, Inc. (SAI), April 1973 to April 1975, McLean, Virginia

I was involved with the design and implementation of two major projects.

The first project was the Atomic Energy Commission's Reactor Safety Study. I was a research analyst involved in developing and applying reliability methods in reactor accident sequence quantification and error/uncertainty propagation. I also was given responsibility for the development of an improved consequence model for the final version of the study.

The second project was the Federal Trade Commission's Market Basket Survey. This survey was designed to statistically determine a "typical" market basket of food for the average family and have an accurate comparison of grocery store pricing. I was retained as an expert consultant to the F.T.C. and helped design and implement the survey and analysis techniques.

Computer Sciences Corporation - August 1970 to April 1973, Arlington, Virginia

I was a task leader with Computer Sciences Corporation where I worked on the general support contract for the National Military Command System Support Center (NMCSSC) in the modeling and gaming department. I designed, implemented, and documented the Data Base Preparation Subsystem of the QUICK Reacting General War Gaming model. I was task leader for the QUICK production support task with responsibilities for maintenance and production support of the model and the associated damage assessment models. I was chosen as War Gaming Analysis Section representative to study and evaluate the consolidation and conversion of the Antiballistic Missile System (ABM-I) and QUICK Strategic War Gaming Models.

Imcor-Glenn Engineering, Inc. - June 1968 to April 1970, Rockville, Maryland

Imcor-Glenn Engineering, Inc. Operations Supervisor, Programmer - I was contracted to work for the Naval Ships Research and Development Center on testing and ev-luation of the Small Boats Project (PCF) and on the Sonar Dome Project. I was also contracted to the Naval Research Laboratory as site team leader for testing and evaluation of Ultra High Frequency Radio Wave Study. As operations supervisor for the Data Division of Imcor, I was responsible for programming and quality control of processed data.

Awards, Honors, and Publications

I received the NRC Special Achievement Award on October 29, 1976 and a NRC High Quality Award on May 11, 1978. I was a session chairman in Consequence Modeling for the American Nuclear Society/European Nuclear Society Topical Meeting on Probabilistic Risk Assessment, September 20-24, 1981 in Port Chester, New York. I was also a session chairman for the American Nuclear

Society Review Conference on the PRA Procedures Guide, April 1982, in Arlington, Virginia. For this conference, I organized three formal debates on current issues in consequence modeling. I have published numerous papers and reports in probabilistic risk assessment, consequence modeling, siting, emergency planning and response, and on the source term. A list of all publications is attached.

Education

I was awarded a Batchelors of Science in Computer Science in 1970 and a Masters of Science in Operations Research in 1973 from the American University in Washington, DC.

AUTHORED OR CO-AUTHORED THE FOLLOWING PUBLICATIONS

"Relationship of Source Term Issue to Emergency Planning," EPRI/NSA Workshop on Technical Factor Relating Impacts from Reactor Releases to Emergency Planning, Bethesda, MD, January 12-13, 1982.

Reactor Safety Study, WASH-1400, Appendix II and VI.

Nuclear Energy Center Site Survey Study, NUREG-001, Exhibit A, Section 6, part IV, "NEC Accident Risk Analysis."

Reactor Accident Source Terms: Design and Siting Perspectives, NUREG-0773, draft.

Regulatory Impact of Nuclear Reactor Accident Source Term Assumptions, NUREG-0771, April 1981.

Task Force Report on Interim Operation of Indian Point, NUREG-0715, August 1980.

Planning Basis for the Development of State and Local Government Radiological Response Plans in Support of Light Water Nuclear Power Plants, NUREG-0396, December 1978.

Emergency Action Level Guidelines for Nuclear Power Plants, NUREG-0610 (Appendix 1 of NUREG-0654, November 1980).

"Consequence Analysis Results Regarding Siting," 1981, Water Reactor Safety Meeting, Gaithersburg, MD.

"Calculations of Reactor Accident Consequences: User's Guide," draft.

A Model of Public Evacuation for Atmospheric Radiological Releases, SAND78-0092, Sandia Laboratories, Albuquerque, NM, June 1978.

Examination of the Use of Potassium Iodide (KI) as an Emergency Protective Measure for Nuclear Reactor Accidents, NUREG/CR-1433, SAND80-0981, Sandia National Laboratories, Albuquerque, NM, March 1980.

"Radiation Protection: An Analysis of Thyroid Blocking," IAEA International Conference on Current Nuclear Power Plant Safety Issues, Stockholm, Sweden, October 20-24, 1980.

"International Standard Problem for Consequence Modeling: Results," International ANS/ENS Topical Meeting on Probabilistic Risk Assessment, Port Chester, NY, September 1981.

"Recent Developments in Consequence Modeling," presented at the Jahreskolloquium PNS, Kernforschungszentrum Karlsruhe, Federal Republic of Germany, November 1981.

"International Standard Problem for Consequence Modeling," International ANS/ENS Topical Meeting on Probabilistic Risk Assessment, Port Chester, NY, September 20-24, 1981.

"Environmental Transport and Consequence Analysis," International ANS/ENS Topical Meeting on Probabilistic Risk Assessment, Port Chester, NY, September 20-24, 1981..

"Weather Sequence Sampling for Risk Calculations," Transactions of the American Nuclear Society, 38, 113, June 1981.

Calculations of Reactor Accident Consequences, Version 2: User's Guide, NUREG/CR-2326, SAND81-1994, Sandia National Laboratories, Albuquerque, NM, (to be published).

"Investigation of the Adequacy of the Meteorological Transport Model Developed for the Reactor Safety Study," ANS Topical Meeting on Probabilistic Analysis of Nuclear Reactor Safety, Newport Beach, CA, May 8-10, 1978.

USNRC, "Environmental Transport and Consequence Analysis," Chapter 9 and Appendices D, E, and F in PRA Procedures Guide, Review Draft, NUREG/CR-2300, 1981.

Overview of the Reactor Safety Study Consequence Model, U. S. Nuclear Regulatory Commission, NUREG-0340, 1977.

PROFESSIONAL QUALIFICATIONS
FRANK H. ROWSOME, 3rd
U.S. NUCLEAR REGULATORY COMMISSION

I am Frank H. Rowsome, 3rd, Deputy Director of the Division of Risk Analysis in the Office of Nuclear Regulatory Research. I have served in that capacity since joining the NRC in July 1979. The work entails planning, budgeting, managing and staffing the Division. Much of the work of the Division is devoted to research in reactor accident risk assessment. The remainder entails risk assessment applied to non-reactor aspects of the nuclear fuel cycle and to standards development related to system reliability or risk.

I received a bachelor's degree in physics from Harvard in 1962. I studied theoretical physics at Cornell, completing all requirements for a Ph.D except for the dissertation in 1965. From 1965 to 1973, I taught and engaged in research in theoretical physics at several colleges and universities.

In 1973 I joined the Bechtel Power Corporation as a nuclear engineer. My initial assignment was to perform accident analyses for nuclear plant license applications. After six months in that job, I was transferred to a newly formed group of systems engineers charged with developing for Bechtel a capability to perform risk assessments and system reliability analyses of the kind the NRC was then developing for the Reactor Safety Study. In that capacity I performed reliability analyses of nuclear plant safety systems, developed computer programs for system reliability analyses, performed analyses of component reliability data, human reliability analyses, and event tree analyses of accident sequences. I progressed from nuclear engineer, to senior engineer, to group leader, to Reliability Group Supervisor before leaving Bechtel to join the NRC in 1979. In this last position at Bechtel, I supervised the application of engineering economics, reliability engineering, and analysis techniques to power plant availability optimization as well as nuclear safety analysis.

While serving as Deputy Director of the Division of Risk Analysis (and its anticedent, the Probabilistic Analysis Staff), I also served as Acting Director (7 months), acting chief of the Reactor Risk Branch (9 months) and acting chief of the Risk Methodology and Data Branch (4 months).

This experience has given me the practitioner's view as well as the manager's view of those facits of reactor risk assessment entailing the classification of reactor accident sequences, system reliability analysis, human reliability analysis, and the estimation of the likelihood of severe reactor accidents. I have the manager's perspective but not the practitioner's experience with those facits entailing containment challenge analysis, consequence analysis, and risk assessment applied to other parts of the nuclear fuel cycle.

My role in the development of testimony for this hearing has been as coordinator of the preparation of testimony on risk and one of the coordinators of the technical critique of the licensee's "Indian Point Probabilistic Safety Study." I am not an expert on the design or operation of the Indian Point plants.

List of Publications

1. "The Role of System Reliability Prediction in Power Plant Design," F.H. Rowsome, III, Power Engineering, February 1977.

2. "How Finely Should Faults be Resolved in Fault Tree Analysis?" by F.H. Rowsome, III, presented at the American Nuclear Society/Canadian Nuclear Association Joint Meeting in Toronto, Canada, June 18, 1976.

3. "The Role of IREP in NRC Programs" F.H. Rowsome, III, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

4. "Fault Tree Analysis of an Auxiliary Feedwater System," F.H. Rowsome, III, Bechtel Power Corp., Gaithersburg Power Division, F 77 805-5.