UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
CONSOLIDATED EDISON COMPANY OF NEW YORK (Indian Point, Unit No. 2)	)	Docket No. 50-247-SP 50-286-SP
	)	
POWER AUTHORITY OF THE STATE OF NEW YORK (Indian Point, Unit No. 3)	)	

DIRECT TESTIMONY OF FRANK ROWSOME AND ROGER BLOND
CONCERNING COMMISSION QUESTION 1

Q.1 Mr. Rowsome, please state your name and business address for the record.

A.2 My name is Frank H. Rowsome, my business mailing address is U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Q.2 Please identify your position with the NRC and describe your responsibilities in that position.

A.2 I am Deputy Director of the Division of Risk Analysis within the Office of Nuclear Regulatory Research. I assist the Director in planning and managing the research program in risk assessment, probabilistic safety analysis, operatons research, reliability engineering, and related regulatory standards development.

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

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

Q.4 Mr. Blond, please state your name and business address for the record.

A.4 My name is Roger M. Blond. My business address is the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

Q.5 Please describe your position with the NRC and describe your responsibilities in that position.

A.5 I am the Section Leader for the Accident Risk Section of the Reactor Risk Branch of the Division of Risk Analysis of the Office of Research. In that position, 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.6 Have you prepared a statement of your professional qualifications?

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

Q.7 What is the purpose of this testimony?

A.7 The purpose of this testimony is to introduce the NRC Staff testimony bearing upon Commission Question One concerning the risk arising from possible accidents at Indian Point Units 2 and 3. We shall outline the risk-related testimony, provide background information about the role of risk assessment techniques, and review the status of NRC Staff studies of risks posed by potential accidents at Indian Point Units 2 and 3.

Q.8 How is the whole of the testimony on risk organized?

A.8 The testimony is organized according to the following outline:

I. INTRODUCTION TO TESTIMONY ON THE RISK-RELATED QUESTIONS

II. Unused heading.*

III. STAFF EVALUATION OF RISK POSED BY INDIAN POINT UNITS 2 AND 3.

A. Accident likelihood

B. Radiological Releases

C. Accident Consequences

IV. SUMMARY RESPONSE TO THE COMMISSION QUESTIONS ON RISK

A. Staff answers to Commission questions on risk

What risks are posed by potential accidents at Indian Point Units 2 and 3 as they were when the IPPSS was done and as they will be in 1983?

B. How will the risk change with --

improvements in Emergency Preparedness?

C. Accuracy of the Staff Risk Predictions .

V. CONTENTIONS AND BOARD QUESTIONS

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* Note that II is an unused heading. (Material prepared for Section II has been merged with I for clarity and continuity)

Q.9 How has the staff interpreted the phrase, "serious accidents... including accidents not considered in the plant's design basis", which appears in Commission Question 1?

A.9 The staff has taken the phrase "serious accident" to be synonymous with severe core damage and core melt accidents. Concerns with the safety of the plant center upon the highly radioactive fission products that accumulate in the reactor fuel as the byproduct of the energy-releasing nuclear reaction. The core of an operating nuclear power plant contains radioactive materials which, if ineffectively contained, can cause substantial harm to workers, and to the population and environment in the vicinity of a plant. Even after an operating reactor is shut down, a mechanism for releasing radioactivity exists. This is called the "afterheat" or "decay heat" produced in the fuel after the nuclear chain reaction ceases. This afterheat diminishes gradually once a nuclear reactor is shut down, but within the first hours or days after shutdown, the afterheat released within the fuel has the potential to melt the fuel and breach each of the several barriers used to obstruct the release of radioactive materials. Such a phenomenon may take place if the afterheat in the fuel is not dissipated in controlled ways. Because a "core melt" accident has the potential to release large quantities of radioactivity, it is the principal cause for concern among potential nuclear reactor accidents.

There is a spectrum of accidents involving the reactor that stop short of severe core damage. The design basis accidents are among

these.* Some of these accidents entail releases of very small amounts of radioactivity, but the releases are far too small to pose a risk to public health and safety, principally because too little of the fission products escape from the fuel to give rise to appreciable doses or contamination offsite. Some of these accidents can do costly damage to plant equipment, but the risk posed by offsite radiological effects is negligible for these accidents which stop short of severe fuel damage. All of the accidents at domestic light water reactors (and at foreign light water reactors of domestic design) that have occurred have been of this very much less type, except for the accident at Three Mile Island.

There are several places in a nuclear plant where radioactive materials are stored in addition to the reactor core. These include the spent fuel storage facilities and the radioactive waste handling and treatment facilities. Studies such as the "Reactor Safety Study" (WASH-1400) and have shown unambiguously that both the risk posed by accidents and the potential hazards from these materials should an accident occur are far lower than for core melt accidents. Thus we can confine our attention to the spectrum of accidents involving severe core damage or meltdown.

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*For purposes of regulatory accident analysis and siting considerations. This is a non-mechanistic assumption. Realistic risk analysis, on the other hand, postulates core damage only in scenarios entailing inadequate core cooling. The design basis accidents, mechanistically analyzed, do not entail severe core damage.

Q.10 What is meant by "risk" in the context of nuclear reactor accidents?

A.10 The technical meaning of risk, as used by the NRC staff, is very much like the definition in common English usage. Risk is a measure of danger that is proportional to both likelihood of accidents and the severity of the consequences. That is, risk is likelihood multiplied by consequences. We speak of distinct measures of risk for each type of undesirable consequence: early death, early injury, delayed (latent) cancer, genetic effects, or property damage.

Mathematically, risk may be portrayed as a graph that displays the severity of the outcome of an accident, e.g., number of casualties, versus the likelihood or frequency of an outcome of at least that severity or greater severity. These graphs are often called "CCDFs," a shorthand term for their formal statistical name: Complementary Cumulative Distribution Functions. A simpler mathematical measure of risk is a single number: the expected value of risk. This is an annual average consequence of accidents. For example, an accident that causes one thousand casualties and occurs once in a thousand years has an expected risk of one casualty per year. So, too, does an accident that occurs once a year and causes one casualty. Thus, the expected value of risk, though simpler than a risk graph like a CCDF, carries less information.

Q.11 What is risk assessment?

A.11 Risk assessment is the discipline of constructing mathematical models that estimate risk. In the context of nuclear reactor accidents, risk assessment entails three principal stages of analysis. In the first stage, accident sequences are identified and the likelihood of occurrence of each sequence is estimated. In the second stage, estimates are made of the damage to the reactor fuel, the success or failure of the containment systems, and the quantity of radioactive materials released to the air and/or groundwater. In the third stage, estimates are made of the consequences beyond the site boundary from the radioactivity released.

Q.12 How are possible accident sequences identified and estimates obtained of the likelihood of each?

A.12 The process of identifying the variety of reactor accidents involving the reactor core begins with catalogs of initiating events, which are disturbances in reactor operation that have the potential to lead to releases of radiation were it not for the intervention or presence of safety systems. Next, the accident sequences are classified according to whether or not systems that can affect the course of the accident operate successfully. Event trees are used in both the qualitative cataloging of accident sequences and in the evaluation of their likelihood. Generally, there is one event tree for each distinct class of initiating event. The event trees display branch points which depict alternative accident sequences. A branch is shown for each system or function whose success or failure leads the incident down a different path.

Thus each route through an event tree depicts a distinct accident sequence.

Next, models of system reliability are constructed. This is commonly done with "fault trees" that trace the origin of system failures to discrete component failures or human errors. A completed fault tree is nothing more than a gigantic sentence in formal logic which can be read: "The system will fail if A happens or if B happens or if C and D happen together or if...," and so forth. These trees catalog the variety of ways a system or group of systems can be disabled. These trees can also be translated into mathematical models which calculate the probability of system failure as a function of the probabilities of the many potentially contributing component failures or human errors.

Finally, the catalogs of initiating events, event trees, and fault trees, are assembled into a mathematical model that permits accident sequence likelihoods to be calculated from initiating event frequencies, component failure rates, and human error rates.

For more detailed information about accident sequences and their probabilities, see Section III.A.

Q.13 What does the containment analysis portion of a reactor risk assessment entail?

A7. The objective of this phase of the risk assessment is to calculate for each accident sequence the timing, quantity, and form of radioactive materials released from the plant. This is done with conventional deterministic (rather than probabilistic) computer-aided calculations of the physical and chemical processes taking place during the evolution of the accident. An analysis is made of the status of core cooling to identify if and when the core overheats. If the analysis predicts core damage or meltdown, then one proceeds to calculate the timing, form and quantity of radioactive material released from the fuel. The slumping of the fuel within the reactor vessel and the attack of the molten core upon the reactor vessel are also analyzed to explore the timing and physical phenomena associated with vessel failure. These analyses generate predictions about releases of steam, energy, and fission products, and about the possibility of missiles from the more violent vessel failure modes. This information is treated as input data to an analysis of the phenomena taking place within the containment building. Further analyses calculate the pressures, temperatures, and composition of the atmosphere within the containment building as the accident evolves. These analyses consider whether or not the containment building remains isolated, whether or not coolers or water sprays are operating within the containment building, whether or not the molten fuel reacts violently with water in the bottom of the reactor cavity, whether or not the molten fuel is cooled there or attacks the concrete floor of the containment building, and so forth. The endpoint of these

calculations is an evaluation of whether or not the containment building fails, how it fails, when it fails, and what materials and energy escape from it to the biosphere.

For more detailed information on containment analysis, see Section III.B.

Q.14 What does the consequence analysis portion of a reactor risk assessment entail?

A.14 The objective of the consequence analysis portion of a reactor risk assessment is to predict the likelihood and extent of early deaths and injuries, delayed deaths and injuries, and offsite property damage caused by reactor accidents. These predictions are made by first calculating the dispersion of radioactive materials released to the open air or subsoil. These calculations are essentially deterministic, although a probabilistic treatment is given to the variety of weather conditions at a site which might prevail at the time of releases. They result in estimates of the likelihood and severity of radioactive exposure at locations outside the reactor site via the air, air-to-ground, or liquid (groundwater) pathways. These calculations result in predictions of the doses of radiation to which members of the population might be exposed at various locations over time. Since these models predict the timing as well as the location of potential doses, a mathmetical model of emergency response can be and generally is employed that accounts for removal of people, either before the radioactive plume arrives or after it passes.

Dose-response relationships are employed to translate calculated doses into effects such as death, radiation illness, cancer, or genetic effects. Assessments of property damage are calculated in terms of the losses sustained in relocating people from contaminated areas, of the interdiction of contaminated foodstuffs, and of post-accident cleanup outside the perimeter of the licensee's property.

Risk assessments do not normally count the losses suffered by the plant owner in the property damage assessments. These losses typically would include lost capital investment in the facility, replacement power, and onsite cleanup.

It is worth noting, however, that these losses by the plant owner are predictably much larger than those suffered by the public due to radiation releases for all but the most extremely severe of potential reactor accidents.

For more detailed information on consequence analysis, see Section III.C.

Q.15 What are the principal strengths and weaknesses of reactor risk assessment?

A.15 Reactor risk assessments derive most of their strengths and weaknesses from the attempt to give a comprehensive, balanced,

realistic model for predicting reactor risks. Probabilistic risk assessment (PRA) is the only known form of reactor safety analysis that can treat multiple component failures, system interactions, and human error in an integrated way. PRA is also unique in providing a comprehensive context for each aspect of reactor accident susceptibility.

The attempt to be comprehensive, however, is bought at a considerable price. The rich variety of root causes, accident sequences, and natural phenomena can only be modeled in a highly approximate way. PRAs are uniquely valuable for their ability to model the integration of very complex phenomena, but the many approximations make PRAs imprecise.

PRAs are constructed on a coherent, logical framework on which is stretched a fabric of numerous, often-simplistic approximations. There are some holes in the fabric, as well. For example, we do not know how to predict the likelihood of sabotage attempts. We have not yet mastered the art of including the contributions to reactor accident susceptibility made by those design errors that are not revealed by either design documents, surveillance tests, or reactor operations. We are not very good at predicting the likelihood that operators might misdiagnose an incident, and so employ the wrong procedures.

Such limitations make PRAs rather unreliable at predicting the precise magnitude of risk. They are, however, very successful at

identifying many--if not all--of the ways a reactor may be vulnerable enough to severe accidents to warrant shutdown or remedial action. They are also extremely valuable as a method with which to estimate the importance of safety issues. A large number of inferences can be drawn from PRAs on ways to improve reactor safety. PRAs also provide a tool with which to evaluate reactor safety issues originating elsewhere even though the issue may not have been modeled in the PRA. PRAs provide an objective framework for putting reactor safety issues in context.

Q.16 What is the history associated with the Commission's investigation into the risk at Indian Point?

A.16 In the spring of 1980 the Nuclear Regulatory Commission was evaluating the actions of the Director of the Office of Nuclear Reactor Regulation with respect to continued operation of the Indian Point Unit 2 and Unit 3 nuclear power plants. As part of that evaluation the Commission asked for the formation of a Task Force of its own staff, with help from others as needed to develop information on the relative risk of the Indian Point power plants and other matters related to consideration of shutting these plants down. The probabilistic analysis staff of the Office of Nuclear Regulatory Research was asked to participate in that Task Force for technical work in risk assessment. The Task Force was formed and instructed by Commission Order on May 30, 1980 to provide insight into the risk posed by the Indian Point power plants compared with the other power plants that were licensed for operation. The Task

Force made its report to the Commission by written report SECY-80-283 on June 12, 1980. This Task Force Report was later published as NUREG-0715 in August 1980. The report NUREG-0715 is adopted as part of this testimony with respect to those portions which deal with accident risks. Some further analyses, not described in SECY-80-283 or in NUREG-0715, were completed after SECY-80-283 and presented to the Commission in oral briefing on June 26, 1980.

Q.17 What occurred after the release of the Reactor Safety Study?

A.17 In the several years following WASH-1400 there was much debate about the probabilistic risk assessment (PRA) methodology that was developed. The debate culminated in the publication in September 1978 of NUREG/CR-0400, the Risk Assessment Review Group Report (Lewis Committee Report). This report was the principal basis for a Commission statement on January 18, 1979, on the use of PRA. Contrary to widespread belief, neither the Lewis Committee report nor the Commission statement disavowed the Reactor Safety Study or the use of PRA. What was disavowed was the short Executive Summary of WASH-1400, which was judged to be an inadequate representation of the Reactor Safety Study. Both the Lewis Committee Report and the Commission statement encouraged careful use of probabilistic risk assessment, especially for setting priorities for regulatory attention. The Three Mile Island (TMI) accident, which came just over a month after the Commission statement, was a graphic example of the need for substantial changes in regulatory emphasis. Probabilistic

risk assessment offers a more rational alternative to understanding the safety significance of a reactor design and site. Since the TMI accident there has been increasing use of PRA in regulatory activities, typical of this was the Commission's specific call for risk analysis in the Indian Point case.

Q.18 What reactors have been analyzed using PRA methods?

A.18 The growing use of PRA since the TMI accident has led to the conduct of many risk studies, which have often interacted with one another even before publication of their results. This interaction is especially important in the case of the Task Force study of Indian Point risk.

The Reactor Safety Study evaluated two reactors as surrogates for the first 100 U.S. power reactors, a Westinghouse 3-loop pressurized water reactor (PWR), Surry, with subatmospheric large dry containment; and a General Electric boiling water reactor (BWR), Peach Bottom, with a Mark I pressure suppression containment. Because of the diversity of reactor design, the NRC initiated the Reactor Safety Study Methodology Applications Program (RSSMAP) to study other reactor types with essentially the same methodology. Four plants were selected: a 4-loop Westinghouse PWR, Sequoyah-1, with an ice-condenser containment; a 2-loop Babcock & Wilcox PWR, Oconee-3, with a large dry containment; a 2-loop Combustion Engineering PWR, Calvert Cliffs-1, with a large dry containment; and a General Electric BWR, Grand Gulf-1, with a Mark III pressure

suppression containment. The resources available did not permit analysis of these plants to the same extent as was done for the Reactor Safety Study so the basic event trees and fault trees of the Safety Study were adapted to these plants based on information obtained from the Final Safety Analysis Report and from some plant visits. Much of the work was concentrated on the analysis of the different containment types. The partial results of the RSSMAP studies which were available in May 1980 played an important part in the Task Force's Indian Point risk assessment. Three of the four RSSMAP reports[1/] have now been published and the fourth[2/] is in final review.

Still another plant risk assessment program has been started by the NRC, the Interim Reliability Evaluation Program (IREP). The first phase of IREP was the study of one plant, a Babcock & Wilcox PWR, Crystal River-3, with a large dry containment. The draft report on Crystal River-3 was available in May 1980 and provided some additional understanding of the important accident conditions for the Indian Point study. The Crystal River-3 IREP report has since been

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1/ NUREG/CR-1659, Reactor Safety Study Methodology Applications Program, Part 1 of 4, Sequoyah #1 PWR Power Plant; Part 2 of 4, Oconee #3 PWR Power Plant; Part 4 of 4, Grand Gulf #1 BWR Power Plant.

2/ NUREG/CR-2515, Crystal River-3 Safety Study, Vols. 1 and 2, January 1982.

published.[3/] The IREP is now in its second phase with a standardized methodology guide and the trial study of four plants nearly complete.[4/] One of the Indian Point plants was to have been included in IREP but the owners of both elected to do a separate, more comprehensive risk study, since IREP focuses principally on systems analysis and the calculation of core melt[5/] probability.

Q.19 What was the Indian Point Short Term Risk Study?

A.19 The purpose of this study was to perform a short-term risk evaluation of the Indian Point 2 and 3 plants. This evaluation was to be used in developing a risk perspective for this high population density site and to identify improvements in design or operation which in the interim have the potential for risk reduction while a more thorough evaluation by the NRC staff and licensee were performed. The short turnaround time allowed for this effort required the use of a simplified approach. As such, the probabilistic risk

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3 The four IREP plants are Browns Ferry-1, Calvery Cliffs-1, Millstone-1 and Arkansas-1. Two have been published.

4/ It should be noted that the analysts who performed the Reactor Safety Study defined core melt as the failure to deliver prescribed core cooling. As the TMI accident demonstrated, degarded core colling can exist for hours without full-scale core melt. Nevertheless, most PRA analysts do not attempt to distinguish severe core damage from core melt and follow the Reactor Safety Study practice of treating degraded core cooling as synonymous with core melt.

5/ The Staff has performed a comparison of the benefits of new mitigation systems in NUREG-0850 and the Licensee has performed a detailed PRA.

estimates obtained are subject to considerable uncertainty, including possible error or inaccuracies. However, the risk perspectives obtained, when combined with good engineering judgment, represent a useful guide to identifying relative risks and potential risk reduction measures.

Q.20 What was the technical approach taken in the study?

A.20 The technical approach taken in this study was to use event tree methodology combined with insights on dominant accident sequences obtained from WASH-1400, RSSMAP, and IREP programs to identify and probabilistically quantify the accident sequences for the Indian Point 2 and 3 designs. These previous risk studies have shown that a handful of accident scenarios would most likely define and dominate a reasonably complete set of core-melt scenarios for a PWR design. Against this experience the Indian Point designs were briefly reviewed. Particular attention was given to identifying common interactions which could affect more than one vital system or could be caused by a single initiating event. The designs were also surveyed for single point vulnerabilities in systems or potential human errors which might significantly influence the likelihood of accident sequences. This information was used in the development of event trees to identify the accident sequences appropriate to the Indian Point plants.

Insights from the previous risk studies and accident sequence analogies were also used in the development of containment failure

mode probabilities and fission product releases. During the course of this brief review, no risk-significant differences between Indian Point 2 nd 3 were identified.

Q.21 What specific event trees were developed in the study?

A.21 Event trees are developed from specific accident initiating conditions. Loss of coolant accidents (LOCA) and transient events (e.g., loss of offsite power) represent the two dominant types of initiating events considered in the Indian Point study. Four event trees were constructed to define the set of accident sequences which could result from significantly different initiating events. Considering the Indian Point design, the LOCA initiators were divided into three event trees, namely: (1) large and intermediate size pipe breaks greater than 2" diameter; (2) small pipe breaks less than 2" diameter; and (3) high to low pressure system interface ruptures (Event V). Transient events analyzed in the study included failure to "scram" and those events which would cause interruption of main feedwater (including loss of offsite power).

Q.22 What was the core melt probability estimate?

A.22 A rough estimate of the overall core-melt probability at Indian Point was made for the plant prior to changes ordered by the Director, NRR, in February 1980. The estimated probability, which has been corrected for minor arithmetic error, is about 4×10^{-5} per year. Due to the February 1980 orders, several improvements to plant operation were identified and a reevaluation of the core melt

probability was made assuming these improvements were incorporated. The revised core melt probability was estimated to be about $1x10^{-5}$ per year.

Q.23 What are the limitations associated with the approach that was taken?

A.23 This short-term risk study of the Indian Point plants has notable limitations because of the methodology used. This study is incomplete; in effect it only looked for dominant risk where dominant risk had been found before in previous assessments of PWRs. If the Indian Point plants suffer from some unique vulnerability which has not been identified before, then this study would not discover it. Thus, this study has a bias which would underestimate the risk of the Indian Point plants.

Accident sequences can also be initiated by external events such as earthquakes, fires, and nearby explosive or toxic chemical hazards. Such accident sequences would be similar in nature to the LOCA and transient sequences; however, it is much more difficult to develop quantitative estimates for the probability of occurrence of external events. It should be noted that the Indian Point short-term risk study analyzed only the risk of internal events. It did not include treatment of external common cause events such as earthquakes and fires because such events were not treated well in WASH-1400 and not treated at all in the RSSMAP and IREP studies. Probabilistic risk analysis of external common cause events is not yet as well

developed as PRA for internal events. Attempts to quantify external events made in previous risk assessment (e.g., WASH-1400, Zion Probabilistic Safety Study) have shown the annual probability of occurrence for such events to be low (i.e., 10-5 - 10-7) in comparison to the probability estimates that have been made for accident sequences associated with LOCAs and transients (i.e., 10-3 - 10-4).

In addition to the above limitations, it must be emphasized that the short-term study of the Indian Point plants took about two man-months of effort for the entire analysis. For comparison, the Reactor Safety Study Methodology Application Program took about one to three man-years of effort per plant and the Interim Reliability Evaluation Program took about eight to ten man-years of effort per plant. The Indian Point Probabilistic Safety Study, IPPSS, has been estimated to have taken 50 man-years of effort. As will be attested to in forthcoming testimony, the IPPSS represents a significant improvement in our level of understanding about the plants' design and operation. Included in the analysis is a comprehensive study of both Units 2 and 3 and a state-of-the art external hazards common cause analysis which harbored the dominant contributors to the risk at the plants.

Q.24 Please provide a brief introduction to the work done by the staff in the last two years to update the analysis of the risks posed by accidents at Indian Point Units 2 and 3 that was presented to the Commission in the summer of 1980 and published as NUREG-0715.

A.24 In the spring of 1980, Harold Denton, Director of the Office of Nuclear Reactor Regulation, charged the staff with preparing an analysis of the desirability of backfits to the Indian Point and Zion plants. The basis for this action was the evidence that these plants may pose a disproportionate share of the societal risks compared with other commercial nuclear power plants by virtue of the comparatively high population density surrounding these plants. The objective of these studies was to determine if retrofits to these plants were warranted to improve the capability of the plants to mitigate the consequences of core melt accidents, in order to reduce the risk so that these plants no longer pose a disproportionate share of the risk - if, in fact, they do.

Thus, the staff embarked on a project to evaluate the effectiveness and reliability with which the containment systems at Indian Point and Zion could bottle up severe reactor accidents, and to evaluate the risk reduction potential associated with a number of hypothetical retrofits including:

(1) filtered, vented containment systems,
(2) combustible gas control systems, and
(3) core retention devices.

The original product envisioned for these studies was a report or series of reports laying the technical groundwork for a regulatory decision on retrofitting the plants.

While these studies were getting under way, the Commission chartered a task force which developed perspectives on the risk posed by the possibility of accidents (the results were published as NUREG-0715) and issued the Memorandum and Order of January 8, 1981 establishing this hearing.

These ongoing studies of accident mitigation were continued with an enlarged scope: they were to constitute the technical basis with which to provide the staff's answer to the Commission questions on risk posed to this board, as well as meet the original goal of evaluating mitigation concepts.

The principal thrust of these studies remained on the mitigation retrofits until the fall of 1981, and culminated in the publication of NUREG-0850, Vol. 1 "Preliminary Assessment of Core Melt Accidents at the Zion and Indian Point Nuclear Power Plants and Strategies for Mitigating Their Effects," November 1981.

Since that time, the studies have been refined, improved, and expanded to address the questions on risk before the board.

These staff studies have focused upon two of the three principal phases of a reactor risk assessment: the containment analysis and the consequence analysis. The staff has developed a position on these two aspects of the risk that is fully independent of the Indian Point Probabilistic Safety Study submitted by the licensees.

The technical material in Section III.B and III.C of this testimony on the containment analysis and on the consequence analysis will describe the staff analyses and also contrast them with the licensees' corresponding analyses.

The remaining principal element of any PRA is the classification of severe accident sequences leading up to a challenge to containment systems and the evaluation of their likelihood. The scope of this portion of a PRA entails the evaluation of the susceptibility of the plant to the possible occurrence of core damage or core meltdown accident. The NRC Office of Nuclear Regulatory Research has considerable experience conducting such studies of a variety of nuclear power plants but has not done such a study of Indian Point, beyond the short term study described in NUREG-0715.

In NUREG-0850, the need for accident likelihood information was filled by the severe accident susceptibility assessment in NUREG-0715 and later improvements on it done in much the same way. That is, it was presumed that the risk-dominant accident sequences at Indian Point were the same as those found in full PRAs of other, similar reactor plants, but the reliability of the systems whose failures give rise to these accident sequences was reevaluated to reflect the specifics of system design at Indian Point. The Staff analysis of the probability of system failure for NUREG-0715 is documented in Appendix 1 of NUREG-0773. The Staff has been well aware that this approach to severe accident susceptibility analysis

might miss vulernabilities not previously highlighted in PRAs of other plants and planned to revise our assessment in the light of the Licensee's study and our critical review of it.

Since the Licensee's study is more comprehensive than the Staff's prior assessment, we have adopted their assessment of core-melt accident sequence likelihoods as a starting point in developing our own.

The Staff contracted with Sandia National Laboratory to critique the accident likelihood portion of the IPPSS and to prepare an improved estimated of the likelihood of severe reactor accidents at Indian Point Units 2 and 3. The Staff has also reviewed both the IPPSS and the Sandia Draft Letter Report.

The Staff has drawn upon the SNL work, our own reviews, and upon the IPPSS in developing our estimates of the likelihood of severe reactor accidents.

Q.25 What is your summary impression of the Indian Point Probabilistic Safety Study?

A.25 The Indian Point Probabilistic Safety Study (IPPSS) is a more comprehensive reactor risk assessment than any published in the U.S. heretofore. It employs approximations that, on balance, are no less conservative than those employed in PRAs done by or for the staff. The licensees deserve a great deal of credit for tackling and publishing this massive and pioneering safety analysis.

The study is noteworthy in that it gives a more thorough accounting than prior PRAs have done of accident initiating events, both those originating in the plant and those due to external events such as earthquake or storms. The Indian Point Probabilistic Safety Study, and the sister study done on Zion, have also broken new ground in the thoroughness with which the challenges to containment by severe reactor accidents have been investigated. So, too, has the staff study of containment challenge phenomena. The IPPSS has also pioneered a technique for the propagation of uncertainties that can accommodate the effects of modeling approximations, phenomenological uncertainties, and completeness as well as the more commonly treated statistical uncertainties.

All the generic limitations of the state of the art in risk assessment described above apply to this study. Many of the approximations that form the fabric of the risk predictive models are known to be pessimistic. Some others are known to be optimistic. For some others, we do not yet know whether the models are conservative or optimistic. We shall know more after our critical review is completed, but the state of the art is such that many of the approximations in the models cannot be unambiguously identified as optimistic or pessimistic.

Q.26 Does this conclude your testimony?

A.26 Yes.

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.

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. Performing and Reviewing Probabilistic Risk Assessments

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. <u>Siting and Emergency Planning and Response Criteria Development</u>

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. Integrating Probabilistic Risk Assessment Into the Regulatory Process

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 evaluation 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.