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 COMPANY OF NEW YORK (Indian Point, Unit 2)	)	50-286-SP
POWER AUTHORITY OF THE STATE OF NEW YORK (Indian Point, Unit 3)	)	

DIRECT TESTIMONY OF FRANK H. ROWSOME
CONCERNING BOARD QUESTION 1.2

Q.1 Please state your name and business address for the record.

A.1 My name is Frank Rowsome. My business address is U.S. Nuclear Regulatory Commission, Washington, D.C. 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, operations 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 What is the purpose of this testimony?

A.4 The purpose of this testimony is to respond to Board Question 1.2, which reads,

> What bearing, if any, do the results reported in NUREG/CR 2497, "Precursors to Potential Severe Core Damage Accidents: 1969-79, A Status Report" (1982), have upon the reliability of the IPPSS? For example, are there specific accident scenarios at Indian Point whose probability may have been inaccurately estimated in light of the real-life data reported and analyzed in NUREG/CR-2497?

Q.5 Why is the precursor study important?

A.5 The precursor study (as I shall call NUREG/CR-2497) suggests that reactor risk assessments performed in the decade of the 70's, such as WASH-1400, may have underpredicted the likelihood of severe core damage accidents. The precursor study also provides a number of perspectives that can be used to illuminate the accuracy of the accident likelihood assessments in contemporary reactor risk assessments such as the Indian Point Probabilistic Safety Study.

Q.6 Please summarize the implications of the precursor study for the accuracy of the IPPSS.

A.6 The precursor study provides no clear indication that the accident likelihood assessment in IPPSS is right or wrong. The precursor study is entirely consistent with the possibility that the IPPSS is correct. The precursor study cannot not rule out the possibility that there might be significant omissions in IPPSS.

The data base, system reliabilities and the qualitative character of severe reactor accidents in the IPPSS seem largely consistent with what one might expect based upon the precursor study. One can, however, find some clues in the precursor study to possible weak spots in the IPPSS. As I discuss below, its is clear that the estimate in the precursor study of the industry-average frequency of severe core damage accidents in the decade of the '70's is unduly pessimistic as a predictor of the frequency of similar kinds of accident sequences at Indian Point.

Q.7 What is the status of the research into severe accident precursor events?

A.7 Oak Ridge National Laboratory, working under contract to the Division of Risk Analysis, Office of Nuclear Regulatory Research, NRC, has been working on accident sequence precursor research since 1979. Science Applications, Inc., working on contract to ORNL on this project, has prepared NUREG/CR-2497, "Precursors to Potential Severe Core Damage Accidents: 1969-1979, A Status Report." We are continuing to support this research. Further improvements in the methodology and the extension of the work to nuclear power experiences since 1979 are ongoing.

Q.8 How are occurrences at nuclear plants distinguished to determine which ones are severe accident precursors and which are not?

A.8 The objective of the screening process is to identify those events that could have contributed directly to the occurrence of a core damage or core melt accident. The screening process entailed preliminary selection of events against six screening criteria, reproduced here as Table 1 followed by technical review of each event that passed the preliminary screen.

Table Selection Criteria for Potential Severe Accident Precursors

Identification of those 1969-1979 LERs that required a detailed review as precursors was made based on an examination of the abstract for each LER. Approximately 19,400 LER abstracts were examined, and specific LERs were chosen if any of the following criteria were met:

1. any failure to function of a system that should have functioned as a consequence of an off-normal event or accident,

2. any instance where two or more failures occurred,

3. all events that resulted in or required initiation of safety-related equipment (except events that only required trip and when trip was successful),

4. all complete losses of offsite power and any less frequent off-normal initiating events or accidents,

5. any event or operating condition that was not enveloped by or proceeded differently from the plant design bases, and

6. any other event that, based on the reviewer's experience, could have resulted in or significantly affected a chain of events leading to potential severe core damage.

Q.9 What bearing does the precursor study have on the accuracy of reactor risk assessments such as the ones on the Indian Point plants?

A.9 Five kinds of results from the precursor study are useful in checking the accuracy of PRAs such as those for Indian Point. First, the precursor report calculates an industry average frequency of occurrence for severe core damage accidents. This can be compared with PRA-based predictions to assess the plausibility of the PRA results. Second, the precursor study lists those events that came closest to being a core damage accident. One can identify if a PRA gave adequate treatment to accident scenarios like these. Third, the precursor study calculates industry-average probabilities of failure of some frequently challenged safety systems. These, too, can be compared with PRA predictions. Fourth, the precursor study lists the causes of system failures for those precursors in which entire safety systems have been found to be disabled. One can verify if the PRA gave adequate treatment to these root causes of failure. Fifth, one can examine the treatment given to those precursor events, if any, that occurred at the plant of interest.

Q.10 How did the precursor study arrive at an industry average frequency of core damage accidents and what results did it show?

A.10 One of the events actually was a core damage accident, the accident at TMI-2. Since the period of the study - the decade of the 70's - included 432 reactor-years of experience, one way of estimating the frequency of core damage accidents in the period is to divide the number of core damage accidents (one) by the number of reactor years. Thus the frequency, f, of core damage accidents can be estimated to have been

$$f = \frac{1}{432} = 2.3 \times 10\text{-}\_ \text{ accidents/reactor year}$$

The precursor study arrived at an estimate of the industry average core melt frequency for the decade of the seventies by doing this calculation and also adding to the number of events other precursors, weighted by an estimate of the conditional probability that each precursor event might have led to a severe core damage accident. In addition, the precursor report added on an additional contribution for those events that would not have been reported in the LER data base.

The result of this calculation is an estimate that the industry average frequency of severe core damage accidents for nuclear power plants in the U.S. during the period of 1969 through 1979 was .0045 accidents per reactor year, i.e., one accident every 222 reactor years. Most of this frequency of accidents was contributed by the three closest calls, the accident at TMI-2, the Browns Ferry fire, and the instrument power supply failure at Rancho Seco. Since reactors have been or are being modified and operators trained to make these three accident types very much less likely in the future, it is interesting to calculate what the frequency of severe core damage accidents would have been if these three events were left out of the calculation. This revised calculation is done in the precursor study. The result is a frequency of 0.000,77 accidents per reactor year, or one accident every 1300 reactor years.

Q.11 In what ways does the precursor study suggest that reactor risk assessments performed in the decade of the 70's, such as WASH-1400, may have underpredicted the likelihood of severe reactor accidents?

A.11 Six plants were the subjects of PRAs in the decade of the 70's. Two are covered in WASH-1400 and four more were studied in the Reactor Safety Study Methodology Applications Program, NUREG/CR-1659. All used essentially the same methodology, assumptions, and data. The core melt frequency estimates in the six studies range from about 10-3 per reactor year to about 10-5 per reactor year. The precursor study, on the other hand, estimates the industry average frequency of severe core damage accidents to have been 4.5x10-3 per reactor year, above the range found in the individual reactor risk analyses.

The difference in the frequency estimates cannot be attributed to the distinction between core melt and severe core damage; the PRA's lumped severe core damage with core melt. Thus, we would expect the frequency estimate in the precursor study should lie somewhere in the range of individual plant frequency estimates. That it does not - and is high - suggests that the PRAs may have underpredicted the severe accident frequency or that the precursor study may have overpredicted the frequency.

Clues to the origin of the apparent discrepancy can be found in the precursor study. 82% of the precursor frequency estimate originates in the three closest calls. The remaining 18%, contributed by all the other precursors combined, agree fairly well both qualitatively and quantitatively with the first six PRAs. Thus, we should examine the treatment of accidents like the three closest calls to find the source of the discrepancy.

The accident at Three Mile Island is the most important of the three; it is the only actual instance of severe core damage.

The PRAs developed before the accident did give a correct qualitative and quantitative treatment of most elements in the TMI accident, except for one important oversight. None of the PRA teams recognized that the peculiar symptoms of a loss of reactor coolant from the reactor's pressurizer could confuse the operators so that they might fail to diagnose that they were dealing with a loss-of-coolant accident.

The PRAs did not identify or treat the possibility that operators might entirely misconstrue an accident in progress, and thus turn off the safety systems that were really necessary to cool the core.

The second most significant precursor was the fire at Browns Ferry. None of the six early PRA's included fire as an accident initiator in the quantitative risk analysis.

The third most significant precursor was the failure of an instrument-and-control power supply at Rancho Seco. None of the six early reactor risk assessments did a very thorough job of exploring the likelihood or fault effects of instrument and control power supply failures. This, too, is a clear cut deficiency in the early PRA's.

Although the remaining precursors are largely consistent with the early PRAs the precursor report does confirm other data studies that suggest that the frequency estimate for small break loss-of-coolant accidents in

WASH-1400 is too low, and that common cause failures of redundant divisions of safety systems due to maintenance errors may be somewhat more likely than the WASH-1400 methodology and data base suggests. The precursor study also identified a few accident initiators other than fire that were not treated in the early PRA's, though none of these contribute significantly to the precursor estimate of accident frequency.

Q.12 How do these predictions of the frequency of severe core damage accidents from the precursor study compare with the predictions for Indian Point?

A.12 The Indian Point Probabilistic Safety Study predictions on the likelihood of severe reactor accidents are compared with the industry-average predictions drawn from the precursor study in Table III 2. On its face, the

TABLE 2

Comparison of the predictions on severe accident likelihood for Indian Point with those developed for the nuclear plant average from severe accident precursors

Plant	Reference	Frequency (per unit year)	Average time to accident (unit years)
Indian Point Unit 2	(1)	4 x 10-4	2,500
Indian Point Unit 2	(3)	1.0 x 10-3	1,000
Indian Point Unit 3	(1)	1 x 10-4	10,000
Indian Point Unit 3	(3)	6.8 x 10-4	1,500
Industry average before fixes	(2)	4.5×10^{-3}	222
Industry average after fixes	(2)	7.7 x 10-4	1,300

Note: Fixes refer to regulatory requirements to reduce the likelihood of accidents such as the TMI accident, the Browns Ferry fire, and the Rancho Seco control fault. Some of these fixes lower the likelihood of other accident scenarios as well, but this has not been considered in the table.

Reference (1) Indian Point Probabilistic Safety Study
(2) NUREG/CP-2497 "Precursors to Potential Severe Core Damage Accidents: 1969-1979 A Status Report" June 1982
(3) NRC Staff estimates for the 1981-82 period when the IPPSS was done. See III of this testimony.

results suggest that the Indian Point plants are safer than average. On the other hand, the differences between the Indian Point predictions and the industry-average inferences from precursors is smaller than the uncertainties in either prediction, so no significance should be attributed to the difference.

The Indian Point plants do appear to be less likely to have severe accidents now than the industry average before the lessons of TMI, the Browns Ferry fire, and the Rancho Seco incident were learned. This difference could originate in one or a combination of factors: (1) the Indian Point units may be genuinely less likely to have severe reactor than the industry average in the decade of the seventies, (2) the IPPSS might be over-optimistic, or (3) the precursor study might be pessimistic. We shall have to look beyond these comparisons of overall accident frequency predictions to resolve these possibilities.

Q.13 Let us turn now to the three dominant precursors found in NUREG/CR-2497. How susceptible are the Indian Point units to the accident scenario that took place at Three Mile Island?

A.13 First, let us consider the specifics of the TMI accident. It is worth noting that even if an Indian Point unit were to be subject to a TMI-like severe core damage or even full meltdown accident, the testimony of Dr. Meyer in Section III.B of this testimony suggests that such accidents would almost certainly be well-contained. Even if such accidents were to have a high frequency at Indian Point, they would contribute very little to offsite radiological risk. However, there are several reasons to believe their frequency is very low. First, plants like Indian Point with Westinghouse reactors very rarely open their pilot operated relief valves in feedwater transients. It was one of these valves that failed to close at TMI causing the loss of reactor coolant. Prior to the fixes inspired by the accident, Babcock & Wilcox reactors routinely opened this valve during feedwater transients. Thus, even before the lessons of TMI had been learned, Westinghouse reactor plants were very much less likely to incur such a loss-of-coolant than were B&W plants. Finally, the operators are now trained and many hardware changes have been made to avoid repetitions of TMI-like accidents. Thus, we can be quite certain that TMI-like accidents contribute negligibly to the risk posed by the Indian Point units.

Q.14 How well does the IPPSS treat errors in diagnosing accidents in progress?

A.14 No PRA, including the IPPSS and this staff testimony, has yet done a thorough job of screening accident scenarios for cases in which the operators might be mislead into a faulty diagnosis of the basic scenario. The analysis of operator error probabilities tends to omit such gross cognitive errors. Operators are far better trained to recognize inadequate core cooling and new instruments have been provided to aid in this disgnosis, so I am confident that operators are far less likely to turn off needed safety systems today than they were before TMI. Still, neither the IPPSS nor the staff testimony has quantified this possibility in a reliable way.

We think such operator errors are quite unlikely today. If this judgment is wrong, then the IPPSS and the staff testimony on risk may have under-predicted the likelihood of severe core damage accidents. Nevertheless, there are two reasons to believe that such omissions - even if present in the IPPSS and the staff calculations - would have very little influence on offsite radiological risk. First, operators are now very well trained to recognize the unmistakable symptoms of severe core damage, such as very high radiation levels in containment. Such scenarios, would almost certainly be nipped in the bud as the accident at TMI was, before core meltdown. Second, even if such scenarios were to lead to full core meltdown, such accidents are predicted to be well-contained. Dr. Meyer's testimony shows that core-melt accidents would be quite reliably contained at Indian Point provided that one of the containment heat removal systems is functioning. There are no scenarios in which the operators would judge it to be desirable to shut off the normally operating containment air coolers. These will suffice to enable the containment to do its job of mitigating the accident.

Q.15 The Browns Ferry fire is ranked as the second most significant precursor in NUREG/CR-2497. What can be inferred about the safety of Indian Point from this event?

A.15 The significance of the Browns Ferry fire for Indian Point lies in the importance of considering in-plant fires in a reactor risk assessment. This was done in the IPPSS and our review of this treatment is described in the testimony of Ben Buchbinder, et al ., in Section III.A above. Note that IPPSS did find that in-plant fires contribute significantly to risk.

Q.16 The Rancho Seco incident in which a Non-Nuclear Instrument bus lost power is ranked as the third most significant precursor in NUREG/CR-2497. What can be inferred about the safety of Indian Point from this event?

A.16 There are several reasons to believe that the Indian Point units are less vulnerable to instrument power supply faults than were some plants like Rancho Seco. The key to the risk significance of the Rancho Seco event lies in the fact that one power supply fault could cause loss of main feedwater, disable or potentially disable the autostart of the emergency feedwater system, and blind the operators to the need to turn on eitner the emergency feedwater system or the emergency core cooling system. The Indian Point units have much better separation of power supplies for safety-related and nonsafety-related instrumentation, and four redundant trains so no one power supply fault could be so disruptive or dangerous.

Although the Indian Point units are less vulnerable to the loss of a single power supply, one might also draw the inference from the Rancho Seco incident that reactor risk assessments should take some pains to develop carefully an analysis of those accidents that might be precipitated by a loss of an auxiliary system such as an instrument power supply. This has been done quite thoroughly for failure in the bulk ac power supplies, i.e., offsite and onsite power. This has also been done for the service water system and the component cooling water system in the IPPSS. Failures of individual dc (control and instrumentation) power supplies were treated in the IPPSS as part of the analysis of the reactor trip event tree. Multiple inverter failures, which reflect failures of DC power supplys, were evaluated in Section 2.4 NUREG/CR 2934, and are best discussed by Messrs. Kolb and Hickman.

Q.17 Please summarize the other important precursors and indicate their significance to Indian Point.

A.17 The remaining precursors altogether contribute only 18% of the predicted frequency of core melt. No one of them stands out as particularly important. Virtually all of them entail either a loss of main feedwater or a loss of offsite power. The assessment of the loss of main feedwater precursors in NUREG/CR-2497 is unduly pessimistic if applied to either Indian Point unit in that the assessment of the likelihood of severe core damage gave no credit to successful core cooling employing the Emergency Core Cooling System, which could be successful in such scenarios at many plants including Indian Point. The assessment of the loss of offsite power precursors in NUREG/CR-2497 is also unduly pessimistic if applied to Indian Point in that it did not credit the possibility in that one of the three gas turbines at the site or the Buchanan Substation could be started and employed to replace offsite power. The IPPSS gives a thorough analysis of both types of accident scenarios.

Q.18 Where can a comparison be found of the industry-average safety system reliabilities estimated in the precursor study and those calculated for Indian Point Units 2 and 3 in the IPPSS?

A.18 The comparison can be found in Section 2.4 of NUREG/CR 2934.

Q.19 Have the root causes of the failure of redundant safety systems identified in the precursor study been given adequate treatment in the IPPSS?

A.19 We have asked Sandia National Laboratory to look into this as part of their review of the IPPSS. The root causes of failures of several systems have been evaluated by Sandia and are discussed in Section 2.4 of NUREG/CR 2934.

Q.20 Did the precursor study identify incidents at either Indian Point Unit 2 or 3 that met the tests to be considered precursor events?

A.20 Yes. One incident at each unit is included in the list of 169 percursor events for the decade of the seventies. Neither was considered to be a significant event according to the precursor study. The loss of offsite power incident of July 13, 1977, with an assessed probability of severe core damage of 5 x 10-4 per occurrence is treated as a precursor because it was an initiating event. However, no failures among the backup systems called upon to respond were recorded. Loss of offsite power incidents are well-treated in the IPPSS.

The other precursor was the heat tracing circuit failure at Indian Point Unit 2 of November 26, 1978. In this incident, the heating system designed to prevent boric acid from crystalizing out of solution in the Boron Injection Tank (BIT) piping failed. This was treated as a precursor for core damage because injection of the contents of the boron injection tank into the reactor coolant system may be necessary in main steam line break accidents. The precursor study determined that there was only one chance in sixteen million that this fault might have contributed to a severe core damage accident. In other words, it was not a remotely close call, and has virtually no significance to the safety of the plant.

There is no suggestion in the precursors attributed to Indian Point Units 2 or 3 that the IPPSS is incorrect, although the experience accumulated at these two plants is too brief to conclude that IPPSS is correct on the basis of precursor event analysis.

Q.21 Are the Indian Point units less likely to have severe accidents today than the precursor report suggests was true of the average plant in the decade of the seventies?

A.21 Yes, it is quite clear that both Indian Point Unit 2 and 3 are less likely to suffer a severe core damage accident of the kinds emerging as signficant in the precursor study than the precursor study suggested was true of the average plant in the 1969-1979 period.

Eighty-two percent of industry average core damage frequency estimate in the precursor study originated from the three closest calls. In the case of the TMI and Rancho Seco scenarios, we have found that the Indian Point units were less susceptible than the plants involved, even before the lessons of these occurrences were learned, and followup changes in design, procedures, and training make them far less likely today. We do not know if the Indian Point units were above or below average in the seventies or, for that matter, today, in their susceptibility to in-plant fires. However, interim actions taken on the basis of the Browns Ferry fire reduced the fire risk.

In addition, the current regulatory program to enforce the new fire protection rule, Appendix R, will still further reduce the susceptibility of the plants to accidents caused by in-plant fires.

The remaining 18% of the industry average core-melt frequency estimate in the precursor study originated almost entirely in loss-of-main-feedwater and loss-of-offsite power precursors. We know that Indian Point Units 2 and 3 are each less susceptible to core damage from such scenarios than the precursor study suggests. There are alternative ways of cooling the core for loss-of-main feedwater events at the Indian Point units not considered in the precursor study. There are also alternate sources of ac power for loss-of-offsite power incidents not shared by other plants or considered in the precursor study.

Thus, it is quite clear that the industry average core damage frequency calculated for the decade of the seventies in the precursor study is an unduly pessimistic predictor of the frequency of such reactor accidents at either Indian Point Unit 2 or 3 today. The precursor report does not, however, shed any light on the likelihood that one of the Indian Point units might fall victim to an earthquake, storm, or other event for which there have been no close calls in power reactor experience. Thus, the precursor study has little bearing on the reliability of the IPPSS.

Q.23 Does this conclude your testimony?

A.23 Yes.

Q.22 Does this conclude your testimony?

A.22 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 prac itioner'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.