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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:
James P. Gleason, Chairman
Frederick J. Shon
Dr. Oscar H. Paris

In the Matter of	
CONSOLIDATED EDISON COMPANY OF NEW YORK, INC. (Indian Point, Unit No. 2)	Docket Nos. 50-247 SP 50-286 SP
POWER AUTHORITY OF THE STATE OF NEW YORK (Indian Point, Unit No. 3)	March 22, 1983

LICENSEES' TESTIMONY OF
THOMAS E. POTTER
ON COMMISSION QUESTION FIVE

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I. INTRODUCTION

My name is Thomas E. Potter. I am a consultant at Pickard, Lowe and Garrick, Inc., in public health consequence analysis of radioactive releases. I was a principal investigator on the Indian Point Probabilistic Safety Study. A statement of my professional qualifications is attached.

This testimony is presented in response to the Board's direction that NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," published by the Nuclear Regulatory Commission (Commission) in December, 1982, be addressed under Question 5. Memorandum and Order at 19 (Nov. 15, 1982). The purpose of NUREG/CR-2239 is to provide technical information for rulemaking on the siting of nucler power reactors. The study was "primarily focused toward the development of generic siting criteria, uncoupled from specific plant design." NUREG/CR-2239 at Foreword.

NUREG/CR-2239 includes the assessment of consequences of hypothetical severe nuclear power plant accidents. For this purpose a "representative" set of three accident releases (SST1, SST2, and SST3) was used to cover the full spectrum of severe light water reactor accidents. The report included "typical" probabilities which might be associated with each of the three releases.

Limitations on the use of the study results to evaluate questions of risks from existing plants are recognized and identified in the foreword to the report:

> There are very large uncertainties associated with these numbers. The absolute values and the ratios of these probabilities for a given facility are design-specific. To accurately portray the risk, very specific accident sequence probabilities and source terms are needed. Thus, the results presented in this report do not represent nuclear power risk.

Id. In other words, the uniform releases and associated frequencies assumed for purposes of NUREG/CR-2239 should not be assumed to apply to real plants. Differences for real plants could affect risk estimates significantly.

The overly conservative emergency response assumptions used in NUREG/CR-2239 also limit the applicability of early fatality risk results to real plants. The authors of the study assumed no emergency response beyond 10 miles for a period of at least 24 hours after passage of the airborne material and acknowledge that peak early fatalities may be overestimated as a result. Id. at 2-51.

Because of these and other limitations, NUREG/CR-2239 cannot be used alone to produce accurate assessments of risks from existing nuclear power plants. The first part of this testimony demonstrates the inapplicability of NUREG/CR-2239 results by comparing it to the licensees' Question 1 testimony[1] showing the effect of differences on risk estimates.

In spite of the limitations described above, NUREG/CR-2239 does include the results of several sensitivity studies relevant to questions raised in this proceeding. These studies are discussed in the second part of the testimony.

II. NUREG/CR-2239 COMPARED WITH INDIAN POINT PLANT-SPECIFIC RISK ASSESSMENT

The major differences between the representative releases used in NUREG/CR-2239 and those determined in the Indian Point Probabilistic Safety Study (IPPSS) are apparent in Table 1. Some IPPSS releases are similar to SST releases, but their frequencies differ. For example, IPPSS release category 2 is similar to SST1, but the frequencies from IPPSS are much lower than the representative frequencies used in NUREG/CR-2239. Likewise, IPPSS release categories 4, 5, 5R, 6, and 7 are roughly similar to SST2, and IPPSS release categories 8A and 8B are similar to SST3, but these release categories are small contributors to risk in both the IPPSS and NUREG/CR-2239.

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1. Licensees' Testimony on Commission Question One Board Question 1.1, and Contention 1.1 (Jan. 24, 1983) (Licensees' Question One Testimony).

TABLE 1

Comparison of NUREG/CR-2239 and
Indian Point Release Categories

Release Type	NUREG/CR-2239 Release Category	NUREG/CR-2239 Frequency (yr^{-1})	Licensees' Question One Testimony Release Category	Licensees' Question One Testimony Mean Frequency (yr^{-1}) IP2	Licensees' Question One Testimony Mean Frequency (yr^{-1}) IP3
Severe release early containment failure	SST1	1.0×10^{-5}	2	4.9×10^{-7}	4.8×10^{-7}
Severe release late containment failure	none	--	2RW	5.8×10^{-5}	1.1×10^{-5}
Other releases (minor contributors to risk)	SST2, SST3	1.2×10^{-4}	all others	1.2×10^{-4}	1.7×10^{-4}

IPPSS release category 2RW has no SST counterpart. This release is characterized by release fracticns somewhat lower than those of IPPSS release category 2 or NUREG/CR-2239 release category SST1, and a very long time between the initiating event and the start of release. In IPPSS this time was assumed to be 12 hours, but most likely would be at least a day. Licensees' Question One Testimony at 24.

The time available for emergency response is very important in limiting early fatality risk. In the IPPSS, release category 2RW was not a contributor to early fatality risk primarily because this time was sufficient for effective evacuation. The NRC Staff analysis (Table III.C-5 evac-reloc) reached the same conclusion for its Release Category RC-C, which is similar to IPPSS 2RW, for evacuation as the selected emergency response. Evacuation assumptions in the Staff analysis[1] were more pessimistic than those in IPPSS. The Staff analysis assumed that evacuation began five hours after the initiating event and that people traveled at 1.5 miles per hour -- a slow walk. Even so, no early fatalities were calculated. In fact, even if the delay were longer and evacuation limited to a zone within six miles of the plant, release category 2RW would still not make a substantial contribution to early fatality risk.

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1. Testimony of Dr. Sarbeswar Acharya Regarding NRC Staff Assessment of Accident Consequences and Risks (Jan. 24, 1983) (Acharya Testimony).

The NRC Staff also analyzed release category RC-C assuming no emergency response for 24 hours after the start of exposure, reasoning that emergency response would be degraded for those releases caused by high winds or earthquakes. When evacuation was assumed, calculations showed no early fatalities. When no emergency response for 24 hours is assumed, means of 493 and 583 early fatalities (conditional on an RC-C release) were calculated for Indian Point Units 2 and 3, respectively. Acharya Testimony at Table III.C-5 (lat reloc.).

The Staff assumption in this case is equivalent to no emergency response for 36 to 48 hours after the initiating event. We believe this is unreasonably conservative. More reasonable assumptions would eliminate RC-C as a contributor to early fatality risk. In addition, Licensees' Testimony on Commission Question One eliminates the release category (Z-1Q) caused by an earthquake-initiated collapse of the containment building; Staff's equivalent release category is RC-A. Accordingly, elimination of release categories RC-C as a contributor to early fatality risk and elimination altogether of release category RC-A would result in reductions in Staff estimates of mean early fatality risk by factors of 25 and 6 for Indian Point Units 2 and 3, respectively. Nonetheless, the comparison of the results of the two Staff analyses dramatically illustrates the importance of time for evacuation in determining early fatality risk.

In the event of severe release following early containment failure -- IPPSS release category 2, NUREG/CR-2239 release category SST1, or NRC Staff release category RC-B -- the effectiveness of evacuation in reducing early fatality risk is limited by the short time available. This explains in large part the NUREG/CR-2239 mean estimate of 830 early fatalities conditional on an SST1 release for Indian Point. See NUREG/CR-2239, Table C-1 (not corrected for power level or other factors). Comparison of the NUREG/CR-2239 SST1 and IPPSS 2RW release category frequencies in Table 1 shows that the use of SST1 with its representative frequency is equivalent to using the SST1 release category (early containment failure) as a surrogate for IPPSS 2RW (late containment failure). Because NUREG/CR-2239 failed to include a release category for late containment failure, the use of NUREG/CR-2239 results without adjustment for plant specific considerations would give large overestimates of early fatality risk at Indian Point. The assumption of early containment failure would be roughly equivalent to an assumption of late containment failure with an unrealistically long delay of 12.5 to 16.5 hours in beginning evacuation.

NUREG/CR-2239 estimates of early fatality risk beyond the 10-mile evacuation zone also limit the applicability of SST1 risk estimates to the Indian Point plants at the Indian Point site.

The IPPSS analysis indicated no early fatalities beyond 10 miles for IPPSS release category 2RW even if there were no emergency response for 24 hours after the passage of the plume. In this respect, release category 2RW differs from release categories IPPSS 2 and NUREG/CR-2239 SST1, primarily because of somewhat lower release fractions and a substantially longer release duration for 2RW, modeled in IPPSS as a multi-puff release. Again, the use of SST1 risk estimates for 2RW would result in substantial overestimate of early fatality risk.

The NUREG/CR-2239 treatment of emergency response beyond 10 miles also limits the value of using SST1 risk estimates as a surrogate for IPPSS release category 2. For NUREG/CR-2239 risk estimates in Appendix C, it was assumed that, for people beyond 10 miles, there was no emergency response for 24 hours after plume passage. For the similar IPPSS release category 2, it was assumed that 90 percent of the population from 10 to 50 miles took shelter equivalent to that provided by the basement of a single family house -- a factor of four reduction in ground dose over that which would be received in normal activities, and a factor of about nine reduction below what would be expected for outdoor exposure. For residents of large apartment buildings, the closing of windows and relocation to rooms away from windows would provide equivalent shielding.

The remaining 10 percent of the population was assumed to carry on normal activities (no shielding other than that normally afforded) allowing for some failures in notification or in gaining access to shelter. For both groups, the exposure period was assumed to be 24 hours. (Shielding is not the only means of achieving dose reductions similar to those used. Early relocation would be as effective.)

This dose reduction assumption was made because such an emergency response would be relatively easily implemented and would be effective. It is inconceivable that such an emergency response would not be implemented in the event of doses approaching life-threatening levels for large numbers of people.

The effect of the assumption is a reduction in the number of early fatalities in the "tail" of the distribution -- the high-consequence, low-frequency end of the curve. This part of the curve results from deposition of substantial portions of the radioactive material released over large ares (tens of square miles) of high population density. The dose-response curve for early fatality (with supportive treatment) is sharp, starting with a probability near zero at about 300 rem to the marrow and rising rapidly to 1.0 at about 600 rem. In these peak cases, doses seldom exceed 600 rem by a large margin so that shelter often eliminates the risks of early fatality. Thus, the IPPSS release category 2 maximum number of early fatalities is about a factor of five lower than that reported in NUREG/CR-2239.

A secondary effect of more realistic emergency response assumptions is a reduction in the number of people needing the full range of supportive treatment to the range of current capabilities for most cases. This does not include all of those persons seeking relief from nausea, which can occur within a day or two following high radiation doses, or those seeking relief from stress-induced illnesses. But it is important to note that the special supportive treatment (such as barrier nursing and transfusions) is not needed at this stage. It need not be started until about one to three weeks after the accident. In the meantime, the affected areas and population could be identified and more elaborate screening implemented prior to treatment.

The general conclusion is that use of NUREG/CR-2239 early fatality conditional risk estimates and representative release frequencies results in an overestimate of early fatality risk from the Indian Point plants at the Indian Point site. It is reasonable to conclude that they are inapplicable to other plant/site combinations as well. Thus, the study results cannot be used alone to perform a comparative risk assessment for existing nuclear power plants.

III. NUREG/CR-2239 SENSITIVITY STUDIES

Although NUREG/CR-2239 cannot be properly used to perform a comparative risk assessment for existing plants, it does include results of sensitivity studies which bear on the question of comparative risk and on other questions raised in the course of this proceeding.

One sensitivity study shows the impact of source term reductions on risk. NUREG/CR-2239 results, reproduced in Table 2, are similar to those reported in the Question One testimony of Dr. William Stratton, Dr. Walton Rodger, and Thomas Potter (Jan. 24, 1983) in this proceeding. All risks decrease with source term reduction, but the decrease is more marked for early health effects. Large reductions effectively eliminate early fatality risk. In the context of risk comparison, it is apparent from results in NUREG/CR-2239 that to the extent that source terms are overestimated early fatality risk is overestimated disproportionately for densely populated sites. Therefore, if source term reductions were uniformly applied to all plants, the impact would be greatest for the higher population density sites. This would also result in lower and narrower ranges of absolute early fatality risk.

A second NUREG/CR-2239 sensitivity study explores the effects of employing varying emergency response strategies. NUREG/CR-2239 at Section 2.5. NUREG/CR-2239 findings

TABLE 2

(Reproduction of Sandia Report Table 2.3.2-2)

Table 2.3.2-2. Sensitivity of Mean Consequences to Reductions in SST1 Release Fractions of All Elements Except Noble Gases[a,b]

Accident Release	Early Fatalities	Early Injuries	Latent Cancer Fatalities	Acute Doses[c] Bone Marrow	Thyroid	Interdicted Land Area
SST1 (Standard)	100[b]	100	100	100	100	100
50% SST1[d]	30	35	74	53	50	55
10% SST1[d]	1	4	32	16	10	10
5% SST1[d]	0.2	2	19	11	5	5
1% SST1[d]	0.03	1	5	8	1	1

a. Assumptions: 1120 MWe reactor, Indian Point Site, New York City meteorology, Summary Evacuation.

b. All consequences normalized to 100 for source term SST1.

c. Relative doses are approximately independent of distance.

d. Release fractions reduced for all isotopes except noble gases.

support and supplement the following conclusions reached in Licensees' Question One Testimony:

- o Emergency response reduces risk for early effects. Delay time is a sensitive parameter for determining effectiveness of evacuation in reducing early fatality risk.

- o Ground dose reduction through the use of either shielding or early relocation is effective in reducing risk for early effects, particularly for the population beyond 10 miles.

- o A strategy of evacuation within 10 miles with shelter from 10 to 25 miles is as effective as evacuation to 25 miles in reducing early fatality risk from the most severe release.

- o Emergency response does not greatly affect latent fatality risk.

The NUREG/CR-2239 conditional risk curves for 91 different sites, reproduced in Figure 1, show that latent cancer fatality risk does not vary greatly from site to site for a given release. Most of the latent fatalities calculated result from exposure to radiation at low dose rates for long periods of time from long-lived isotopes deposited at low concentrations over large areas. The latent fatality risk depends upon the population density within the affected area. However, for large areas, differences in population density among sites are relatively small. This phenomenon is noted by the authors of NUREG/CR-2239 in discussing conditional cancer fatality risk: "Thus, the distributions

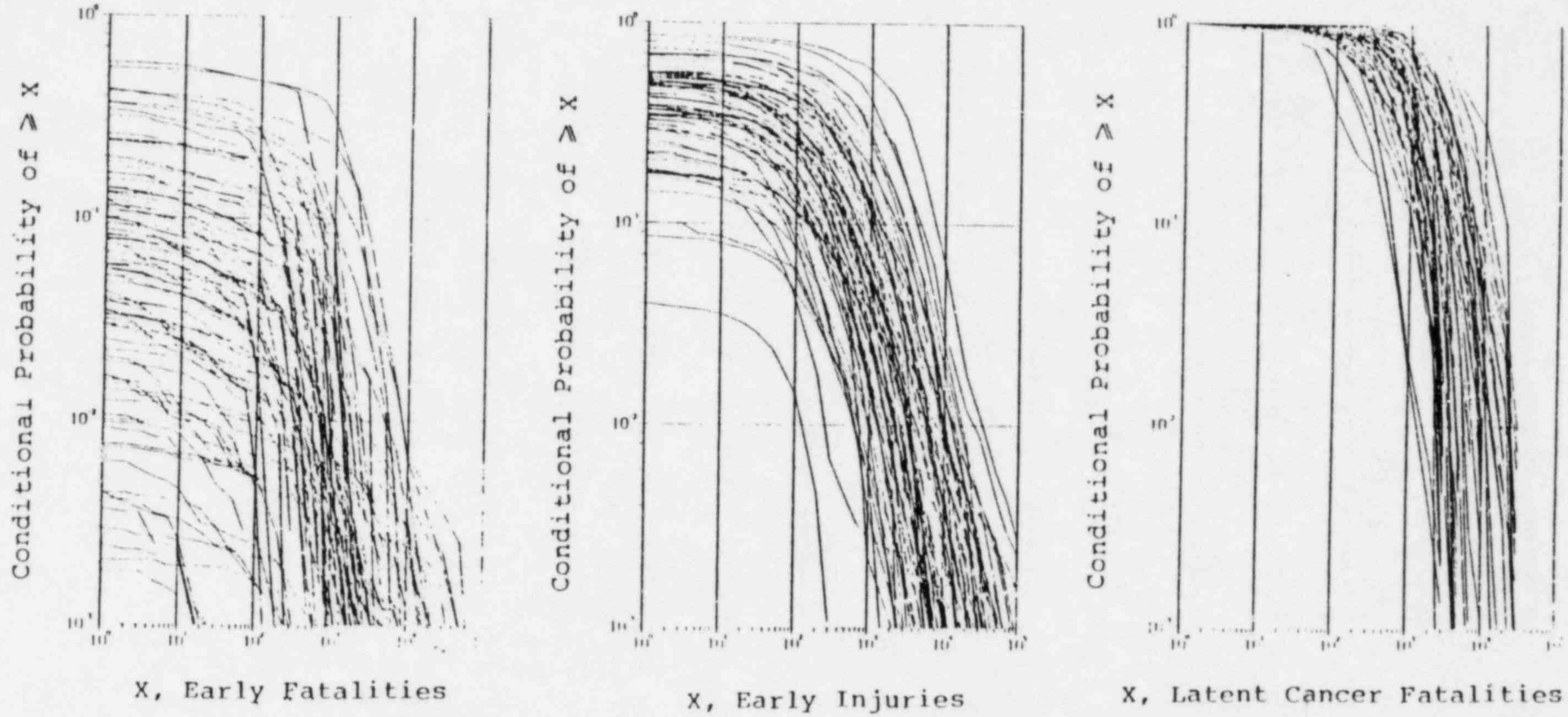

FIGURE 1

Figure 2.4.2-1. (a) Early Fatality, (b) Early Injury, and (c) Latent Cancer Fatality CCDFs Conditional on an SST1 Release at all 91 Current U.S. Reactor Sites. Assumptions: 1120 MWe reactor, Summary Evacuation, representative meteorology. Range of means: early fatalities 0.4 to 970, early injuries 4 to 3600, and latent cancer fatalities 230 to 8100.

of latent cancer fatalities, which can occur over very large areas, show the least variability." NUREG/CR-2239 at 2-37.

Accordingly, many nuclear power plants in the northeast and Great Lakes regions of the United States have similar latent fatality risks. This is because major population centers fall within the latent fatality risk zone of each. The ratio of the Indian Point mean latent fatality risk to the average mean latent fatality risk for all sites (for SST1) is 4.7. This ratio is conservative because the Indian Point Unit 3 power level (965 MWe) is approximately 15 percent below the power level assumed in NUREG/CR-2239 (1120 MWe). The Indian Point Unit 2 power level (873 MWe) is approximately 22 percent lower than the NUREG/CR-2239 assumed power level. Additionally, of the 165 reactors listed in the report, 91 (55 percent) have power levels higher than Indian Point. Considering these two factors, correcting NUREG/CR-2239 Table C-1 for power level would result in a ratio even lower than 4.7. This suggests that the variation of severe release frequency from plant to plant may be at least as important as the variation in population density from site to site in making comparative latent fatality risk estimates for real plants at real sites.

NAME

THOMAS E. POTTER

EDUCATION

M.S., Environmental Science, University of Michigan, 1972.
B.S., Chemistry, University of Pittsburgh, 1963.

PROFESSIONAL EXPERIENCE

General Summary

Consultant on health and safety aspects of nuclear power. Performing environmental dose assessments for nuclear power plant safety analysis, environmental reports and operating reports. Assisting clients in design and implementation of radiological or environmental monitoring programs and interpretation of results. Providing independent review of in-plant radiological protection programs and effluent analysis programs.

Consultant in radiological health aspects of nuclear power. Prepared radiological health section of safety analysis reports and environmental monitoring programs and evaluated data from those programs. Developed a mathematical model to predict radiation doses from nuclear power plant effluents.

License administrator, plutonium fuel facility health and safety supervisor. Provided radiological safety review of major facility modifications. Used these analyses and nuclear criticality analyses perfomed by others to prepare AEC special nuclear materials and byproduct license applications. Served as corporate contact with AEC in matters related to licensing. Organized and supervised a radiological protection program for a plutonium fuels fabrication facility and hot cell facility. Instituted personnel monitoring programs using thermoluminescent dosimetry and breathing-zone areosol sampling in 1967. Served as secretary of a plant safety committee which inspected all operations and reviewed detailed written procedures for operators. Served as member of a corporate safety committee which determined corporate policy regarding health and safety matters.

Chronological Summary

1973-Present	Consultant, Pickard, Lowe and Garrick, Inc.
1972-1973	Consultant to Dr. G. Hoyt Whipple, University of Michigan.
1963-1970	Nuclear Materials and Equipment Corporation (NUMEC). License administrator, plutonium fuel facility health and safety supervisor.

MEMBERSHIPS

American Chemical Society.
American Nuclear Society.
Health Physics Society.
Certified by American Board of Health Physics.

REPORTS AND PUBLICATIONS

Woodard, K., and T. E. Potter, "Consideration of Source Term in Relation to Emergency Planning Requirements," presented to the Workshop of Technical Factors Relating Impacts from Reactor Releases to Emergency Planning, Bethesda, Maryland, January 12-13, 1982.

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