DOCKETED
USNRC

'83 MAR 24 A10

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) POWER AUTHORITY OF THE STATE OF NEW YORK (Indian Point, Unit No. 3)	Docket Nos. 50-247 SP 50-286 SP March 22, 1983

LICENSEES' TESTIMONY OF DENNIS C. BLEY, DONALD F. PADDLEFORD, THOMAS E. POTTER, AND DENNIS C. RICHARDSON ON COMMISSION QUESTION FIVE

ATTORNEYS FILING THIS DOCUMENT:

Brent L. Brandenburg

CONSOLIDATED EDISON COMPANY
OF NEW YORK, INC.
4 Irving Place
New York, New York 10003
(212) 460-4600

Charles Morgan, Jr.
Paul F. Colarulli
Joseph J. Levin, Jr.

MORGAN ASSOCIATES, CHARTERED
1899 L Street, N.W.
Washington, D.C. 20036
(202) 466-7000

8303250149 830322
PDR ADOCK 05000247
T PDR

TABLE OF CONTENTS

I. PRESENTATION AND QUALIFICATIONS OF PANEL MEMBERS

My name is Dennis C. Bley, Ph.D. I am a consultant at Pickard, Lowe and Garrick, Inc., in reliability, risk, and decision analysis for electrical generating plants. I was a principal investigator on the Indian Point Probabilistic Safety Study. A statement of my professional qualifications is attached.

My name is Donald F. Paddleford. I am an Advisory Engineer in the Risk Assessment Section of the Nuclear Safety Department of the Nuclear Technology Division of Westinghouse Electric Corporation. I was a principal investigator on the Indian Point Probabilistic Safety Study. A statement of my professional qualifications is attached.

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.

My name is Dennis C. Richardson. I am the Risk Assessment Technology Manager in the Nuclear Safety Department of the Nuclear Technology Division of Westinghouse Electric Corporation. I was a principal investigator on the Indian Point Probabilistic Safety Study. A statement of my professional qualifications is attached.

II. INTRODUCTION

A central issue in this hearing is whether the Indian Point nuclear power plants produce risks that significantly exceed the range of risks posed by other nuclear power plants in light of the demographic characteristics of the area surrounding the Indian Point site. This issue is articulated in Question 5 of the Commission's Memorandum and Order of January 8, 1981, which asked:

> Based on the foregoing, how do the risks posed by Indian Point Units 2 and 3 compare with the range of risks posed by other nuclear power plants licensed to operate by the Commission? (The Board should limit its inquiry to generic examination of the range of risks and not go into any site specific examination other than for Indian Point itself, except to the extent raised by the Task Force.)

Risk can be measured by several health and economic indices and from both an individual and a societal standpoint. Population distribution and plant characteristics affect these indices differently. In selecting which indices are most important, guidance is taken from the Nuclear Regulatory Commission's (Commission's) preliminary safety goals, which emphasize early and latent fatality risks (Reference 1). Similarly, the emphasis here is on the effects of population distribution on early and latent fatality risk.

Three different approaches to addressing Commission Question 5 are taken in this testimony. First, a comparison is made of the risks from the Indian Point plants to the Commission's preliminary safety goals. Second, the risks from the Indian Point plants, as analyzed in the Indian Point Probabilistic Safety Study (IPPSS) (Reference 2), are compared to the results of site and plant specific probabilistic risk assessments (PRAs) of a number of other nuclear power plants. Third, there is a discussion of the benefits resulting from the special design features at Indian Point which are not present at all nuclear power plants.

Individually and collectively, each of these comparisons supports the conclusion that the Indian Point nuclear power plants are in the range of risks posed by other nuclear power plants. Specifically, (1) the risk of latent fatalities at Indian Point is low and information available suggests that latent fatality risks may not vary greatly among nuclear power plants; (2) the absolute risk of early fatalities is even lower than the latent fatality risk, thereby reducing the significance of plant-to-plant variability; (3) for both risk indices, the Indian Point plants meet the Commission's preliminary safety goals; and (4) anticipated reductions in source term estimates would reduce both early and latent fatality risk and, in fact, could effectively eliminate the early fatality risk. <u>See</u> Licensees' Testimony of William R. Stratton, Walton A. Rodger, and Thomas E. Potter on Question One (Jan. 24, 1983). In addition, plant features present at Indian Point but not included at other plants are among the important factors supporting the conclusion that the Indian Point nuclear power plants are within the range of risks posed by other nuclear power plants.

III. COMPARISON WITH THE NUCLEAR REGULATORY COMMISSION'S PRELIMINARY SAFETY GOALS

On March 14, 1983, the Commission published a Policy Statement on Safety Goals for the Operation of Nuclear Power Plants. 48 Fed. Reg. 10,772 (1983). The preliminary safety goals and design objectives apply to both individual and societal risk. Subordinate to these goals is a design objective for risk to the plant, core melt frequency.

The preliminary safety goals represent a national benchmark against which all nuclear power plants can be compared. Therefore, the comparison of the risks from the Indian Point plants to the Commission's preliminary safety goals is one method of determining if these plants are within the range of risks posed by other nuclear power plants. Both Indian Point Units 2 and 3 are among those plants which have health risks smaller than those adopted by the Commission's preliminary safety goals.

Uncertainties in the calculated health risks for Indian Point are offset by the large margins between these calculated risks and the preliminary safety goals. Reduced source terms will result in even larger margins.

A. Individual Risk

The Commission's preliminary safety goals state that the early fatality risk to an average individual in the vicinity of a nuclear power plant[1] should not exceed one-tenth of one percent of the sum of early fatality risk to that individual from other accidents. 48 Fed. Reg. 10,774.

To translate this goal into numerical form, we use the United States national average accident risk of 5 fatal accidents per 10,000 people per year (5×10^{-4} per year) (Reference 2).

For the purpose of assessing the individual risk, the Commission defines "vicinity" of the plant as a 1-mile radius. 48 Fed. Reg. 10,774. Using this definition of vicinity and IPPSS emergency response assumptions, the average individual early fatality risk has been calculated

---

1. According to the Commission,

> the average individual in the vicinity of the plant is defined as the average individual biologically (in terms of age and other risk factors) and locationally who resides within a mile from the plant site boundary. This means that the average individual is found by accumulating the estimated individual risks and dividing by the number of individuals residing in the vicinity of the plant.

48 Fed. Reg. 10,774.

as a fraction of the national average accident risk. This is then compared with the Commission's preliminary safety goal in Table 1. The risk of Indian Point is well within this goal, by a factor of approximately 70 for Indian Point Unit 2 and a factor of approximately 75 for Indian Point Unit 3.

B. Societal Risk

For societal risk, the Commission's preliminary goal is that the latent cancer fatality risk to the population in the vicinity of a nuclear power plant should be less than one-tenth of one percent of the cancer fatality risks from other causes. 48 Fed. Reg. 10,774. For latent fatalities, vicinity is defined as 50 miles. Id. The national average cancer risk for a person in the United States is two deaths per 1,000 people per year (2×10^{-3} per year) (Reference 2).

For this radius from the Indian Point plants, the average latent cancer fatality risk has been calculated as a fraction of the national cancer fatality risk and is compared with the Commission's preliminary goal in Table 1. The risk of Indian Point is well within this goal, by a factor of approximately 165 for Indian Point Unit 2 and 710 for Indian Point Unit 3.

C. Core Melt Frequency

Table 1 also shows the comparison of the Indian Point Units 2 and 3 median core melt frequencies with the Commission's preliminary safety goal. Because the Zion PRA

TABLE 1

COMPARISON OF RISKS FROM
INDIAN POINT PLANTS WITH NRC SAFETY GOALS

	Indian Point 2	Indian Point 3	NRC Goal
Average Early Fatality Risk Within 1 Mile as a Fraction of Other Accident Fatality Risk Within 1 Mile	1.4 x. 10^{-5}	1.3×10^{-5}	1×10^{-3}
Average Latent Cancer Fatality Risk Within 50 Miles as a Fraction of Other Cancer Fatality Risks Within 50 Miles	6.0×10^{-6}	1.4×10^{-6}	1×10^{-3}
Core Melt Frequency (per year of reactor operation) internal plus external	1.4×10^{-4}*	5.0×10^{-5}*	1×10^{-4}
Core Melt Frequency (per year of reactor operation) internal only	5.0×10^{-5}*	3.0×10^{-5}*	no explicit goal stated

*Median Frequency

(Reference 3) and the IPPSS are the only risk assessments of which we are aware that give comprehensive treatment to external events, Table 1 also includes the median core melt frequency of internal initiating events only. Considering internal initiating events only, both Indian Point plants meet the Commission's preliminary safety goal.

Although information on core melt frequency is provided here for completeness in comparing the Indian Point plants against the preliminary goals, the values of this parameter are not of particular use in addressing Commission Question 5. This is because core melt frequency is a poor indicator of public risk, as discussed in Licensees' Testimony on Commission Question One, Board Question 1.1, and Contention 1.1 (Jan. 24, 1983). This can be shown in two ways. First, approximately 65 percent of the postulated core melt scenarios at Indian Point Unit 2 and almost 95 percent of those at Indian Point Unit 3 do not lead to significant releases of radioactive material to the environment. Second, approximately 95 percent of the calculated early fatality risk at each plant is due to the interfacing systems LOCA, which contributes less than one-half of one percent to the core melt frequency. On the other hand, core melt frequency is a useful indicator of economic risk to the customers and owners of Indian Point Units 2 and 3, as it is a measure of the likelihood of losing the benefits of these plants.

IV. PRA COMPARISONS

Another way to compare the risks posed by the Indian Point plants with those posed by other nuclear power plants is to compare site and plant specific PRAs for various plants, all identical in scope and using state-of-the-art methodology. At the present time, however, such a comparison cannot be made due to the limited number of available, comparable studies. The following comparisons, however, are possible:

1. A comparison of the IPPSS risk results with those of other plants for which reasonably complete PRAs have been published. Only the Indian Point and Zion PRAs include external events; therefore, only comparisons on an internal initiator basis have been made.[1]

2. A comparison of the IPPSS results with the range of risks for nuclear plants calculated by the Commission Task Force Report on the Interim Operation of Indian Point (Reference 4).

A. Comparison Of Risks Among Nuclear Power Plants

In connection with the comparison of risks among nuclear power plants, it is important to note that PRA methodology has been evolving rapidly over the last 10 years. The various published studies, therefore, differ considerably in certain respects. Thus, when comparing the

---

1. While the Big Rock Point PRA did consider fires, it did not evaluate other external initiating events.

results of IPPSS with those of other PRA studies, it must be recognized that such comparisons are not only of different plants, but are also of different data bases and, in some cases, of different methodologies. These studies vary in scope and sophistication. Some did not include external events and/or public health effects, while others focused only on a few systems or on one type of accident initiator. With these reservations in mind, quantitative comparisons can be made.

1. Individual Risk

Table 2, which was compiled by the Commission Staff (Reference 5), presents data from a number of plant specific PRAs on the frequency of core melt, the frequency of a major release, and the early and latent fatality risks to an individual living within one mile of plant boundaries. The values in the "Early fatality" column can be directly compared with the Commission's preliminary safety goal for this health index (5×10^{-7}). This table generally reflects the range of risks from internal initiating events at United States nuclear power plants because it includes a representative sampling of PWRs and BWRs, high and low population density sites, power levels from 71 to 1250 MWe, and principal reactor vendors and architect engineers. Although this table has been reproduced verbatim from Reference 5, additional information is also presented for Indian Point Units 2 and 3, and appears in a box directly below the

TABLE 2

RESULTS OF EXISTING PROBABILISTIC RISK ASSESSMENTS

*** WARNING - THERE ARE LARGE UNCERTAINTIES ASSOCIATED WITH THE VALUES PRESENTED IN THIS TABLE. ALSO, PRAs WERE NOT PERFORMED USING CONSISTENT METHODOLOGY AND ASSUMPTIONS

	PRA	NSSS/AE	DATE/POWER (MWe)		F CORE MELT 1/ 2/	F MAJOR RELEASE 1/ 3/	INDIVIDUAL RISK WITHIN 1 MILE: EARLY FATAL 1/ 4/	INDIVIDUAL RISK WITHIN 1 MILE: CANCER FATAL 1/ 5/	COMMENTS
ANO-1	IREP	B&W/BECHTEL	81	836	5×10^{-5}	2×10^{-5}	6×10^{-7}	2×10^{-7}	all-PWR 2 8/
Biblis B	GERMAN RSS	FRG (W)	78	1300	4×10^{-5}	1×10^{-6}	3×10^{-8}	2×10^{-8}	Containment stronger and larger than U.S.
Big Rock 6/	WOOD-LEAVER/ SAI	GE/BECHTEL	81	71	1×10^{-3}	0	0	---	Low power level, remote siting 9/
Browns Ferry	IREP	GE/TVA (BWR 4, M I)	81	1067	2×10^{-4}	4×10^{-5}	2×10^{-7}	1×10^{-6}	ATWS and interdependency in redundant RHR trains dominate core melt 10/
Calvert Cliffs	RSSMAP	CE/BECHTEL	82	850	2×10^{-3}	1×10^{-3}	9×10^{-6}	2×10^{-5}	More comprehensive IREP study in progress. AFWS redesign will lower risk and core melt frequency 11/
Crystal River	IREP	B&W/GILBERT	80	825	4×10^{-4}	2×10^{-4}	3 x 10-6	2×10^{-6}	P core melt reduced by factor of 3 by procedure changes 12/
Grand Gulf	RSSMAP	GE/BECHTEL (BWR 6, M III)	81	1250	4×10^{-5}	4×10^{-5}	1×10^{-7}	1×10^{-7}	Containment always fails directly to atmosphere, does not assume staff's analysis of ATWS risk
I.P. #2 6/	PLG	W/UE&C	82	873	4×10^{-4}	3×10^{-4}	3×10^{-8}	1×10^{-8}	Includes external events 7/ 13/
	PLG	W/UE&C	83	873	1×10^{-4}	4×10^{-8}	7×10^{-9} a	6×10^{-8} a	Includes external events
	PLG	W/UE&C	83	873	5×10^{-5}	4×10^{-8}	6×10^{-9} a	3×10^{-9} a	Internal events only
	a. This is a mean value. The median value would be somewhat lower.								
I.P. #3 6/	PLG	W/UE&C	82	965	9×10^{-5}	3×10^{-5}	1×10^{-9}	3×10^{-10}	Includes external events 7/
	PLG	W/UE&C	83	965	5×10^{-5}	4×10^{-8}	6×10^{-9} a	2×10^{-8} a	Includes external events
	PLG	W/UE&C	83	965	3×10^{-5}	4×10^{-8}	6×10^{-9} a	3×10^{-9} a	Internal events only
	a. This is a mean value. The median value would be somewhat lower.								
Limerick 6/	SAI	GE/BECHTEL (BWR 4, M II)	81	1055	2×10^{-5}	3×10^{-6}	1×10^{-8}	1×10^{-6}	Mean value, assumes ATWS fix
Millstone	IREP	GE/EBASCO	82	652	3×10^{-4}	1×10^{-4}	1×10^{-7}	6×10^{-7}	Major release is in Release Category 3
Oconee	RSSMAP	B&W/BECHTEL	80	860	8×10^{-5}	4×10^{-5}	2×10^{-7}	1×10^{-7}	1/4-PWR 2; 3/4-PWR 3
Peach Bottom	WASH-1400	GE/BECHTEL (BWR 4, M I)	75	1065	3×10^{-5}	7×10^{-6}	4×10^{-8}	3×10^{-8}	Staff's analysis of ATWS would likely result in risk exceeding safety goal
Sequoyah	RSSMAP	W-IC/TVA	78	1148	6×10^{-5}	4×10^{-5}	1×10^{-6}	5×10^{-7}	H_2 control reduces risk by 2 to 3
Surry	WASH-1400	W/S&W	75	775	6×10^{-5}	1×10^{-5}	2×10^{-7}	1×10^{-7}	2/3-PWR 2; 1/3-PWR 3
Zion 6/	PLG	W/S&L	81	1100	4×10^{-5}	4×10^{-6}	2×10^{-8}	1×10^{-8}	Includes external events

1/ All numbers are median values or point estimates from internal initiators unless otherwise specified.

2/ Frequency of core melt 1×10^{-4} is the Safety Goal Value for Accident Probability Comparison.

3/ Frequency of release with Potential for Early Fatalities Assuming Nominal Evacuation and Warning Times (RSS).

4/ 5×10^{-7} is the Safety Goal for Early Fatality Risk Comparison. Same assumptions as 3 above unless specified.

5/ 2×10^{-6} is the Safety Goal for Cancer Fatality Risk Comparison. Same assumptions as 3 above unless specified.

6/ Utility-performed PRAs. All values are rough estimates based upon initial interpretation of results.

7/ Optimistic emergency response assumptions (1-hour delay with at least 8-hour warning) for dominant sequence when determining individual risk.

8/ Predicted risk is dominated by small LOCAs and transients. Souce term reduction expected to reduce predicted risk to within guidelines. Likelihood of major release could be reduced by adding parallel valves at the discharge of the borated water storage tank or by improving DC power redundancy.

9/ Low power level (71 MWe) results in low individual risk. Extensive design modifications necessary to reduce core melt frequency.

10/ Reduction of core melt frequency would require redesign of the residual heat removal system to eliminate commonalities between trains which reduce the significance of multiple redundancy.

11/ AFWS redesign is expected to significantly reduce core melt frequency and individual risk. IREP study including improved AFWS design will be available in Spring 1983. Modification to DC power system and engineered safety system actuation system may be required to lower core melt frequency within guidelines. Predicted risk is dominated by transient event and should be significantly reduced by new source term data.

12/ Core melt frequency could be reduced to less than guidelines levels by improving written procedures and improving the reliability of the steam supply to the EFWS turbine driven pump. Predicted risk is dominated by small LOCA events. New source term information is expected to result in a moderate reduction in predicted risk.

13/ Core melt frequency is dominated by seismic considerations.

14/ Core melt frequency could be reduced to below guideline levels by redesigning the emergency AC power system to reduce dependency on the gas turbine and improving procedures for responding to transients. Predicted risk is dominated by transient events. New source term information should result in a significant reduction.

Indian Point results presented by the Staff. This additional information is drawn from the risk calculations in Licensees' Testimony on Commission Question One, Board Question 1.1, and Contention 1.1. It includes risk results from internal initiating events only to avoid an erroneous comparison of Indian Point internal plus external results with internal only results from other plants. It also includes the internal plus external results for Indian Point. Based on the results in this table, the risk to an individual living within 1 mile of Indian Point compares favorably with the estimated risk to individuals living within 1 mile of other nuclear power plants. Additionally, the Indian Point core melt frequency is within the range of other estimates presented in the table, and the frequency of a major release compares favorably with the estimates for the other plants in the table.

Another valuable comparison is the frequency of the interfacing systems LOCA, which is believed to be an important contributor to early fatality risk at PWRs and is the major initiating event contributing to the early fatality risk at Indian Point. Estimates of the frequency of this event at Indian Point and several other nuclear power plants are presented in Table 3. The differences in these estimated frequencies are due to a combination of design differences among plants, as well as to testing and

TABLE 3

COMPARISON OF INTERFACING SYSTEMS LOCA MEDIAN FREQUENCIES

Study	PWR Plant	Median Frequency	Recurrence Interval (Number of Reactor Years)	Reference
RSS	Surry	4×10^{-6}	250,000	6
IPPSS	Indian Point 2	4×10^{-8}	25,000,000	2
IPPSS	Indian Point 3	4×10^{-8}	25,000,000	2
ZPSS	Zion	3×10^{-8}	33,000,000	3
RSSMAP	Oconee	7×10^{-5}	14,000	7
RSSMAP	Sequoyah	5×10^{-6}	200,000	7
IREP	Crystal River-3	2×10^{-9}	500,000,000	8

maintenance procedures. The IPPSS accounted for testing and maintenance, including some procedures which are not in effect at all other plants.

2. Societal Risk

Societal risk comparisons have been compiled for the PRAs listed in Table 4. Graphical comparisons of results from these studies are presented in Figures 1 and 2. Many of the studies listed in Table 2 did not calculate risk curves and are, therefore, not included in Figures 1 and 2. The results from the German Biblis B risk study are included in these figures, as in the Staff table, because the study is recent, reasonably comprehensive, and analyzes a high population site.

Because so few PRA studies have comprehensively examined external initiating events as does IPPSS, the comparisons in these figures are for internal initiating events only. (The risk curves presented in the licensees' Question 1 testimony included both internal and external events.) Figure 1 shows the median risk curves for early fatalities as presented in the various studies, and Figure 2 presents similar results for latent cancer fatalities. These figures support the conclusion that Indian Point is within the range of societal risks posed by other nuclear power plants.

TABLE 4

PLANTS USED IN THE GRAPHICAL COMPARISONS

Plant	Reference
Indian Point 2	2
Indian Point 3	2
Surry	6
Peach Bottom	6
Zion	3
Biblis B	9
Big Rock Point	10
Limerick	11

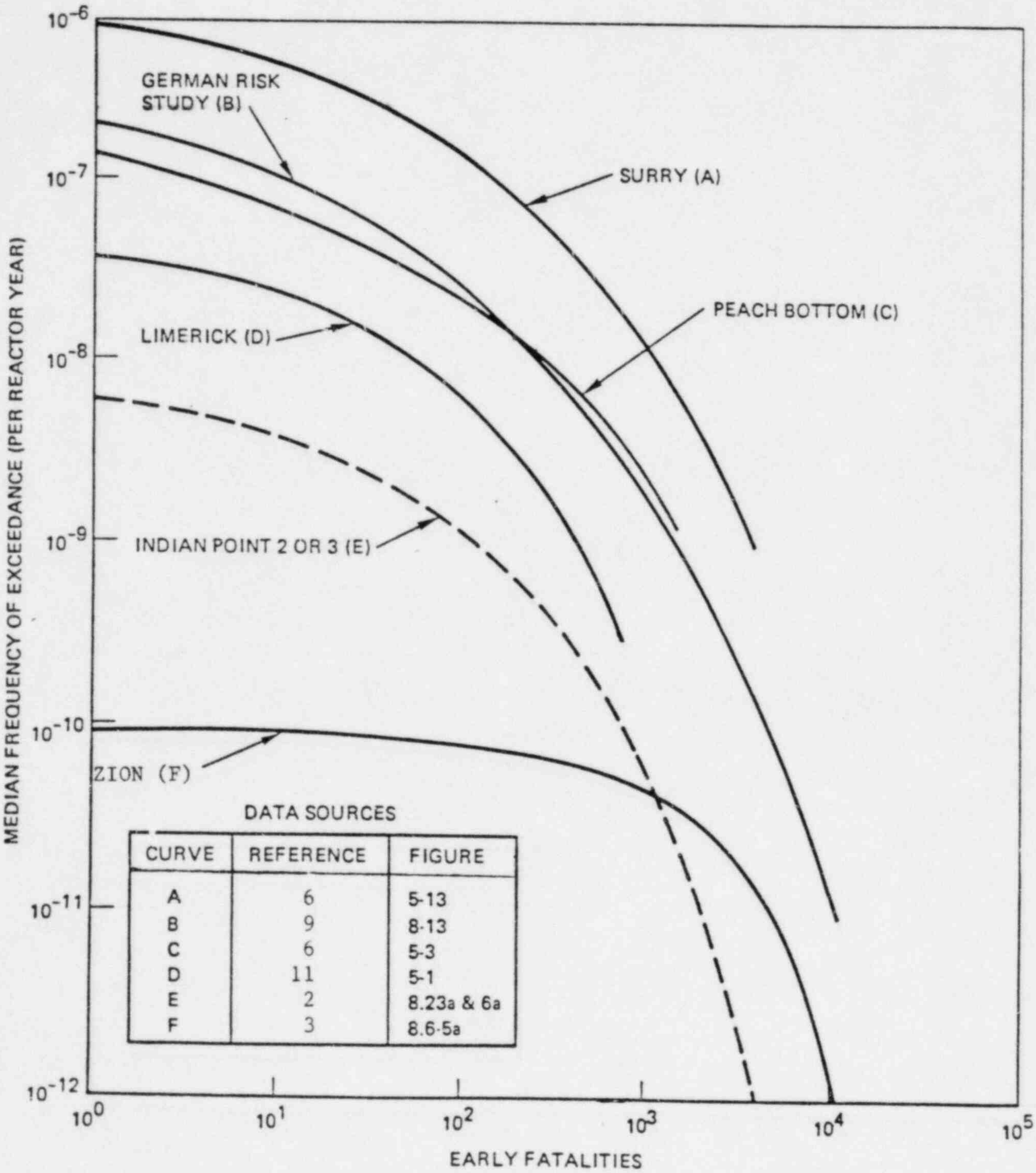
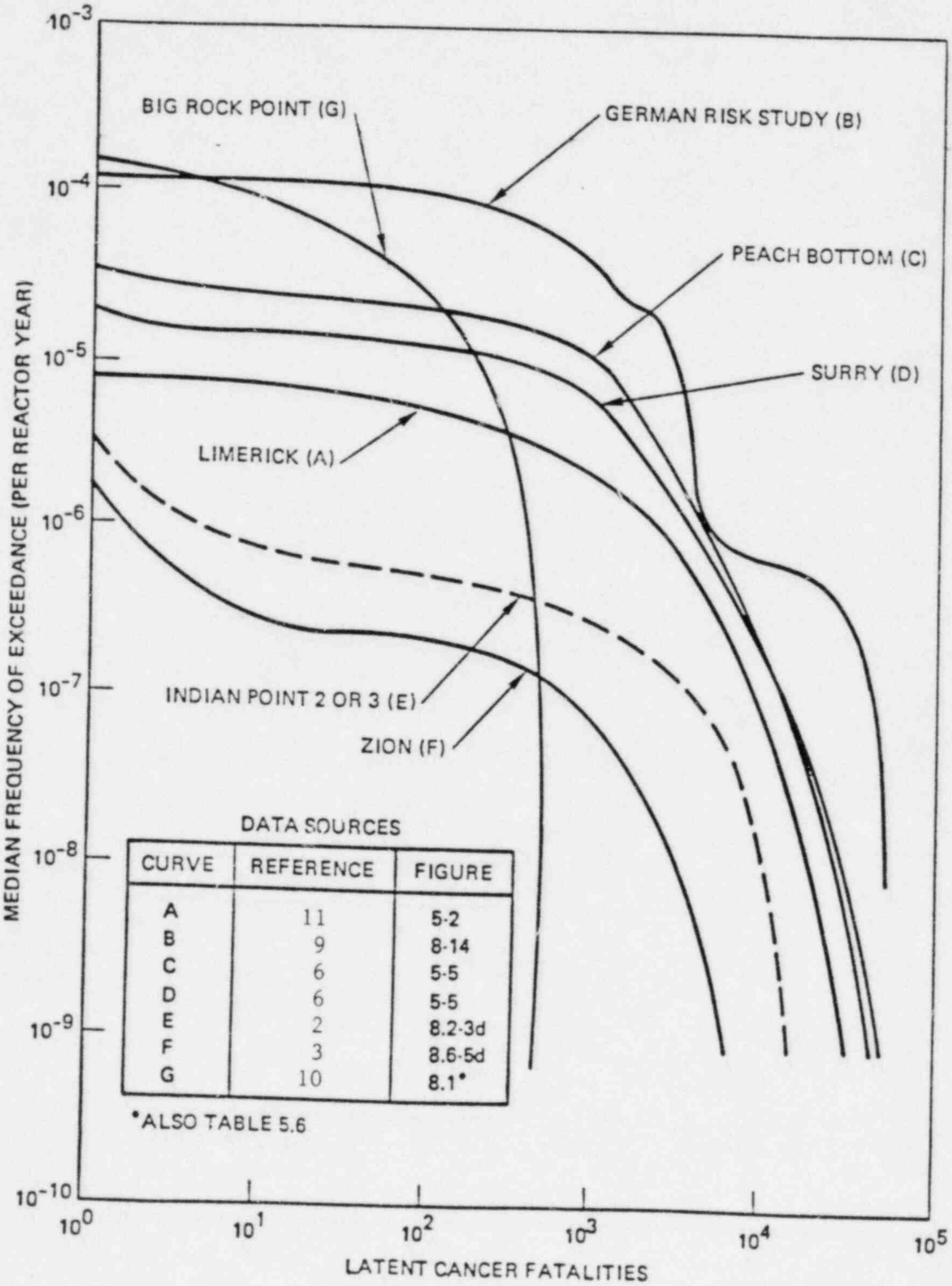


Figure 1. Comparison of PRA Median Risk Curves
for Early Fatalities (Internal Risk Only)



CURVE	REFERENCE	FIGURE
A	11	5-2
B	9	8-14
C	6	5-5
D	6	5-5
E	2	8.2-3d
F	3	8.6-5d
G	10	8.1*

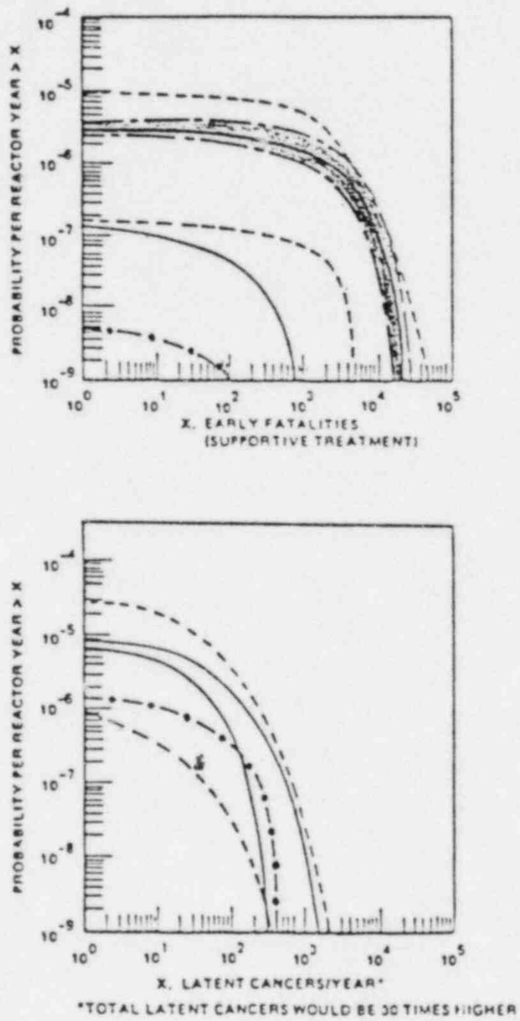
Figure 2. Comparison of PRA Median Risk Curves for Latent Fatalities (Internal Risk Only)

B. Comparison with the Commission's Task Force Results

In 1980, a Commission task force studied the effects on risk of: (1) a typical pressurized water reactor (Reactor Safety Study, Surry) at different sites; (2) different plants at the same site (Indian Point); and (3) different public protection measures (Reference 4). Because the results of these studies are an indication of the range of risks posed by nuclear plants in general, they are also used for the comparison requested in Commission Question 5.

For this purpose, the median internal risk curves from the IPPSS are presented in Figure 3 along with results from Figure 11 of the Task Force Study for early and latent fatality risk. These curves support the view that the risk from Indian Point is within the range of risks from other nuclear power plants.

As can be seen from Figure 3, the early fatality risk curve calculated in the IPPSS lies more than an order of magnitude below the range of results presented in the Task Force Study. A large part of this difference results from the Task Force Study's failure to evaluate the strength of the containment, which precludes prompt containment failure. Thus, the Task Force did not include a release category for late containment failure. The IPPSS latent fatality risk curve lies within the range of the latent fatality risk calculated by the Task Force for the Indian Point site, and is below the range calculated for the Surry reactor at various sites.



NOTE 1. THE RANGES REPRESENT BEST ESTIMATES ON A COMPARATIVE BASIS. THERE ARE LARGE UNCERTAINTIES WITH THE ABSOLUTE VALUES PRESENTED IN THIS FIGURE.

2. PUBLIC PROTECTIVE MEASURES HAD NO SIGNIFICANT IMPACT ON TOTAL LATENT CANCER

ESTIMATED RANGE OF CONSEQUENCES FOR VARIOUS DESIGNS CONSIDERED AT INDIAN POINT SITE.*

ESTIMATED RANGE OF CONSEQUENCES FOR SIX SITES CONSIDERED WITH SURRY DESIGN.*

ESTIMATED RANGE OF CONSEQUENCES FOR VARIOUS PUBLIC PROTECTIVE MEASURES CONSIDERED AT INDIAN POINT SITE.*

INDIAN POINT 2 OR 3 – INTERNAL ONLY.**

*REFERENCE 1.
**REFERENCE 2.

Figure 3. Ranges of Risk Variation

V. SPECIAL DESIGN FEATURES AT INDIAN POINT

The comparisons presented in the previous sections indicate that the Indian Point Units are within the range of risks of other nuclear power plants, despite the demographic characteristics of the area surrounding the Indian Point plants. It is thus appropriate to ask whether this conclusion is supported by information about the engineering and design features of the plants.

Nuclear power plants located at high population sites have received special attention from regulatory agencies. During the licensing review for the Indian Point Units, additional features were incorporated into the plant designs to supplement the standard safety features. These features were highlighted in the Director's Order of February 11, 1980.

Among the features that could lead to lower frequencies of major releases from the Indian Point containments than from some other containments are:

(1) The design and construction of these containments, with a pressure limit of 141 psia and a large volume of 2.6×10^6 cu. ft., gives them the capability to withstand internal pressures well in excess of the design pressure of 62 psia. Additionally, the containments can withstand without significant structural damage all credible seismic events that could occur in this area. The containment building configuration allows gases to circulate and mix easily to prevent local accumulation of hydrogen. This configuration also provides for more effective containment heat

removal capability. In addition, the geometry of the reactor cavity promotes dispersion of the core debris, thereby increasing its coolability. Also, the geometry of the containment floor provides for easy entry of water to the reactor cavity to cool the debris.

(2) Containment cooling capability is provided by diverse systems. The design includes five fan cooling units in addition to four pumps capable of providing containment spray recirculation. The availability of any one of the fans or sprays is sufficient to prevent containment overpressure failure. Two recirculation pumps, located inside containment, are unique to Indian Point and are two of the pumps capable of providing containment spray.

(3) The Indian Point containments have two sumps that provide for recirculation of emergency core cooling water. The presence of two sumps is also unique to Indian Point.

(4) The presence of the recirculation pumps inside containment provides the capability of recirculating emergency core cooling water without its leaving the containment building.

(5) Three gas turbine-generators are available for supplying power to either unit. This feature is unique to Indian Point and provides an unusual degree of diversity in emergency power sources.

(6) Confirmatory signals (S signals) are sent upon actuation of emergency safeguards to certain power operated isolation valves to ensure that, if a valve had been inadvertently placed in an incorrect position, it would be restored to its correct position. This feature reduces the likelihood of bypassing the containment.

(7) The containment weld channel pressurization system and the isolation valve

seal water system help to assure that the containment leaktightness is maintained.

(8) The service water and component cooling water systems are arranged to maximize redundancy of active components. Any one of six service water pumps can supply any service water load. Similarly, either of two component cooling water pumps can be connected to any component cooling water load. The flexibility provided by these and similar interconnections within and between systems results in particularly low risk from internal initiating events at Indian Point.

The risk reductions afforded by some of the design features discussed above have been quantified using information from the IPPSS. For example, the frequency of late overpressure containment failure from internal initiating events is reduced by one to two orders of magnitude by the presence of fan coolers, which back up the spray recirculation system. The gas turbines, an additional source of AC power recovery for the time period of one to three hours following a core melt, provide up to an order of magnitude reduction in the frequency of late overpressure containment failures from internal initiating events. When external as well as internal initiating events are considered, the fan coolers provide up to an order of magnitude reduction and the gas turbines provide less than a factor of two reduction in the frequency of late overpressure containment failures. While not specifically quantified, the other design features discussed above would certainly provide further risk reduction.

On the strength of these special design features and other specific Indian Point systems, less than 2 percent of the internally initiated core melts lead to containment failure. As indicated in Table 2 and supported here, the frequency of a major release resulting from internal initiating events is thought to be less at Indian Point than at a number of other nuclear power plants. In addition, as stated above, the various safety features, particularly the fan coolers, provide significant reductions in overall (internal plus external) frequency of late containment failure.

As discussed in Licensees' Testimony of Thomas E. Potter on Commission Question Five (Mar. 22, 1983), the range of latent fatality risk among nuclear power plant sites, given a severe release, is relatively narrow. Based on the information in Table 2, the strength of the Indian Point containments, and the special design features at the plants, the release frequency at Indian Point is lower than the estimated release frequencies at many other plants. The narrow range of latent fatality risk, in conjunction with a lower than average release frequency, supports the conclusion that the Indian Point latent fatality risk is within the range of latent fatality risk of other nuclear power plants.

Based on the information in Table 2, the absolute value of the early fatality risk at a number of nuclear power plants is very low. At Indian Point, this is largely due to the strength of the containments, which essentially precludes prompt containment failure. The only accident contributing to early fatality risk is the interfacing systems LOCA which, as shown in Table 4, has a very low frequency of occurrence.

Special design features, together with standard nuclear power plant safety systems, result in very low early and latent fatality risk at Indian Point Units 2 and 3.

VI. CONCLUSIONS

Each of the several comparisons used in this testimony to address Commission Question Five supports the conclusion that Indian Point Units 2 and 3 are within the range of risks posed by other nuclear power plants.

A comparison of the Indian Point plants to the Commission's preliminary safety goals shows that these plants are within these goals. As such, they are in the class of plants whose risks are in a range below the limits established by these goals.

Various comparisons of the results of other PRAs to the results of the IPPSS show that the Indian Point plants are within the range of risks estimated for other nuclear power plants. Table 2 indicates that the early fatality risk for a number of nuclear power plants, including Indian Point, is very low. The Indian Point early fatality risk is low because, based on the strength of the containments, the low frequency interfacing systems LOCA is the only contributor to early fatality risk at Indian Point.

Using the source terms proposed in the previously submitted Question 1 testimony of Dr. William Stratton, Dr. Walton Rodger, and Thomas Potter, no early fatalities would occur for any Indian Point accident scenario.

When absolute risks are very low, differences between these low numbers are relatively unimportant.

With regard to the latent fatality risk, the Indian Point plants are close to the national average of the mean values of latent fatality consequences, based on the generic work reported in NUREG/CR-2239 (Reference 12). This report shows that the range of latent fatality consequences, given a specified release, is relatively narrow. See Licensees' Testimony of Thomas E. Potter on Commission Question Five (Mar. 22, 1983).

Based on the strength of the Indian Point containments and the special features of the Indian Point plants, radioactive releases from these plants would be less frequent than at many other plants. See Table 2. The narrow range of the consequences and the lower frequency of containment failure support the conclusion that the latent fatality risk is within the range of such risks posed by other nuclear power plants.

The above conclusion on latent fatalities is also relevant to the issue of whether any mitigation devices are warranted for the Indian Point plants. As discussed under Commission Questions 1 and 2, the principal application of these mitigation devices would be to reduce latent fatalities. The Indian Point plants have latent fatality risks which meet the Commission's preliminary safety goals and are within the range calculated for other nuclear power plants. This range itself will be lower and narrower with reductions in source terms. Therefore, no additional mitigation features are necessary to bring Indian Point within the range of risks posed by other nuclear power plants.

REFERENCES

1. "Policy Statement on Safety Goals for the Operation of Nuclear Power Plants," 48 Fed. Reg. 10,773 (1983).

2. Consolidated Edison Company of New York, Inc., and the Power Authority of the State of New York, "Indian Point Probabilistic Safety Study" (Mar. 1982), and Amendment 1 (Jan. 1983).

3. Commonwealth Edison Company, "Zion Probabilistic Safety Study" (Sept. 1981).

4. NUREG-0715, "Report of the Task Force on Interim Operation of Indian Point," SECY-80-283 (1980).

5. Memorandum from William J. Dircks to the Commission (Draft) (Jan. 5, 1983) (Attachment).

6. WASH-1400 (NUREG/75-014), "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants" (1975).

7. NUREG/CR-1659, "Reactor Safety Study Methodology Applications Program" (1981).

8. NUREG/CR-2515, "Crystal River-3 Safety Study, Main Report," Vol. I (1981).

9. EPRI-NP-1804-SR, "German Risk Study - Main Report. A Study of the Risks Due to Accidents in Nuclear Power Plants" (1981) (English Translation).

10. Consumers Power Company, "Probabilistic Risk Assessment, Main Report, Big Rock Point Plant" (1981).

11. Philadelphia Electric Company, "Limerick Probabilistic Safety Study" (1981).

12. NUREG/CR-2239, "Technical Guidance for Siting Criteria Development" (1982).

NAME

DENNIS C. BLEY

EDUCATION

Ph.D., Nuclear Reactor Engineering, Massachusetts Institute of Technology, 1979.
Courses in nuclear engineering and computer science, Cornell University, 1972-1974.
U.S. Navy Nuclear Power School, 1968.
University of Cincinnati, B.S.E.E., 1967.
Courses in Mathematics and Physics, Centre College of Kentucky, 1961-1963.

PROFESSIONAL EXPERIENCE

General Summary

A consultant at Pickard, Lowe & Garrick, Inc., 1979-present. Technical analysis of power plant availability and risk. Cost-benefit analysis of power plant system changes. Preparation of technical reports, expert testimony, and proposals. Supervision of the technical quality of PLG reports and direction of some PLG projects. Instructor at availability, risk, and decision analysis courses offered by PLG. Oyster Creek Probabilistic Risk Assessment (OPSA). Assisted in the completion and review of this complete risk assessment of an operating BWR performed for Jersey Central Power & Light. Work Order Scheduling System (WOSS). Assisted in developing the San Onofre 2 and 3 plant model for a computer based work order prioritizing, scheduling, and record keeping system for Southern California Edison Company. Steam Turbine Diagnostics Cost-Benefit Analysis. Developed and applied a procedure for evaluating diagnostic alternatives for EPRI. Reliability Analysis of Diablo Canyon Auxiliary Feedwater System for Pacific Gas & Electric. Midland Plant Auxiliary Feedwater System Reliability Analysis for Consumers Power. Technical Review of the "Office of Emergency Services Recommended Emergency Planning Zone Considerations..." for Southern California Edison. Prioritization of NRC Action Plan for NSAC. Development of a methodology and participation in an AIF workshop to apply it for EPRI/NSAC. Zion and Indian Point Probabilistic Safety Studies. Methods development, systems analysis, and plant modeling. Other PRAs--LaSalle, Browns Ferry, Midland, Pilgrim 1, and Oconee.

On USS Enterprise, Reactor Training Assistant, 5 months, 1971. Responsible for technical training of approximately 400 nuclear trained officers and men prior to annual safeguards examination. Propulsion Plant Station Officer, 9 months, 1970-1971. Responsible for maintenance and operation of one propulsion plant (two reactors, eight steam generators, and associated equipment) during power range testing of new reactors and during deployment. Approximately 50 enlisted personnel were assigned to the plant. Shift Propulsion Plant Watch Officer, 15 months, 1969-1970. Supervised a crew of about 20 navy enlisted operators and many shipyard workers on 8-hour shift rotation conducting maintenance and testing in one propulsion plant during refueling-overhaul. Shipboard qualifications: Propulsion Duty Officer, responsible for all propulsion equipment during absence of Reactor Officer and Engineer Officer. Engineering Officer of the Watch, operational watch in Central Control, responsible for all propulsion and engineering equipment and watch standers. Propulsion Plant Watch Officer, operational watch in one propulsion plant, directed and responsible for all operations in the plant.

At Cincinnati Bell, Plant staff assistant, 4 months, 1967. Worked in central office and transmission group supplying technical assistance to the line organization. Cooperative trainee, 3 years, 1964-1967, work-study program with alternate three month periods at the University of Cincinnati.

Chronological Summary

1979-Present	Consultant, Pickard, Lowe and Garrick, Inc.
1974-1979	Massachusetts Institute of Technology. Research assistant for Department of Energy LWR Assessment Project. Teaching assistant in engineering of nuclear reactors.
Summer 1976	Northeast Utilities. Engineer: economy studies, plant startup, analysis of physics tests.
1967-1974	U.S. Naval Reserve, active duty. Instructor of naval science, Cornell University, 1971-1974; Reactor Department of USS Enterprise, deployment and refueling-overhaul, 1969-1971; Nuclear Power training program and Officer Candidate School, 1967-1969.
1964-1967	Cincinnati Bell. Plant staff assistant and work-study program trainee.

MEMBERSHIPS, LICENSES, AND HONORS

The Society for Risk Assessment.
Institute of Electrical and Electronics Engineers.
American Nuclear Society.
American Association for the Advancement of Science.
The New York Academy of Sciences.
U.S. Naval Reserve, Commander.
Registered Nuclear Engineer, State of California.

Sigma Xi (national science honors society), 1976.
Sherman R. Knapp Fellowship (Northeast Utilities), 1975-1976.
Sloan Research Traineeship, 1974-1975.
Eta Kappa Nu (national electrical engineering honors society), 1967.

REPORTS AND PUBLICATIONS

"Seabrook Probabilistic Safety Assessment," Public Service Company of New Hampshire, to be published in 1983.

Pickard, Lowe and Garrick, Inc., "Midland Probabilistic Risk Assessment," Consumers Power Company, to be published in 1982.

Oconee Probabilistic Risk Assessment," a joint effort of the Nuclear Safety Analysis Center, Duke Power, and other participating utilities, to be published in 1982.

Tennessee Valley Authority and Pickard, Lowe and Garrick, Inc., "Browns Ferry Probabilistic Risk Assessment," to be published in 1982.

Apostolakis, G., M. Kazarians, and D. C. Bley, "A Methodology for Assessing the Risk from Cable Fires," accepted for publication in Nuclear Safety, 1982.

Kaplan, S., H. F. Perla, and D. C. Bley, "A Methodology for Seismic Safety Analysis of Nuclear Power Plants," proposed presentation at the International Meeting on Thermal Nuclear Reactor Safety, Chicago, Illinois, August 29-September 2, 1982.

Bley, D. C., S. Kaplan, and B. J. Garrick, "Assembling and Decomposing PRA Results: A Matrix Formalism," proposed presentation at the International Meeting on Thermal Nuclear Reactor Safety, Chicago, Illinois, August 29-September 2, 1982.

Garrick, B. J., S. Kaplan, and D. C. Bley, "Recent Advances in Probabilistic Risk Assessment," prepared for the MIT Nuclear Power Reactor Safety Course, Cambridge, Massachusetts, July 19, 1982.

Fleming, K. N., S. Kaplan, and B. J. Garrick, "Seabrook Probabilistic Safety Assessment Management Plan,"PLG-0239, June 1982.

Garrick, B. J., "Lessons Learned From First Generation Nuclear Plant Probabilistic Risk Assessments," to be presented at the Workshop on Low-Probability/High-Consequence Risk Analysis, Arlington, Virginia, June 15-17, 1982.

Garrick, B. J., S. Kaplan, D. C. Iden, E. B. Cleveland, H. F. Perla, D. C. Bley, D. W. Stillwell, H. V. Schneider, and G. Apostolakis, "Power Plant Availability Engineering: Methods of Analysis, Program Planning, and Applications," EPRI NP-2168, PLG-0165, May 1982.

Bley, D. C., and R. J. Mulvihill, "Comments on Evaluation of Availability Improvement Options for Moss Landing Units 6 and 7," PLG-0226, March 1982.

Stillwell, D. W., G. Apostolakis, D. C. Bley, P. H. Raabe, R. J. Mulvihill, S. Kaplan, and B. J. Garrick, "EEI Availability Handbook," PLG-0218, January 1982.

Bley, D. C., L. G. H. Sarmanian, and D. W. Stillwell, "Reliability Analysis of Safety Injection System Modification, San Onofre Nuclear Generating Station - Unit 1," PLG-0206, October 1981.

"Zion Probabilistic Safety Study," Commonwealth Edison Company, September 1981.

Buttemer, D. R., "Analysis of Postulated Accidents During Low Power Testing at the San Onofre Nuclear Generating Station--Unit 2," PLG-0199, September 1981.

Bley, D. C., D. W. Stillwell, and R. R. Fray, "Reliability Analysis of Diablo Canyon Auxiliary Feedwater System," presented at the Tenth Biennial Topical Conference on Reactor Operating Experience, Cleveland, Ohio, August 17-19, 1981.

Garrick, B. J., and D. C. Bley, "Lessons Learned from Current PRAs," presented to the ACRS Subcommittee on Reliability and Probabilistic Risk Assessment, Los Angeles, California, July 23, 1981.

Kaplan, S., G. Apostolakis, B. J. Garrick, D. C. Bley, and K. Woodard, "Methodology for Probabilistic Risk Assessment of Nuclear Power Plants," draft version of a book in preparation, PLG-0209, June 1981.

Perla, H. F., "Project Plan: Probabilistic Risk Assessment, Midland Nuclear Power Plant," PLG-0180, May 1981.

Bley, D. C., C. L. Cate, D. W. Stillwell, and B. J. Garrick, "Midland Plant Auxiliary Feedwater System Reliability Analysis Synopsis," PLG-0166, March 1981.

Pickard, Lowe and Garrick, Inc., "A Methodology to Quantify Uncertainty of Cost of Electricity for Alternate Designs of (Combustion) Turbine Combined Cycle Plants," PLG-0162, March 1981.

Garrick, B. J., S. Ahmed, and D. C. Bley, "A Methodology for Evaluating the Costs and Benefits of Power Plant Diagnostic Techniques," submitted for presentation at the Ninth Turbomachinery Symposium, Houston, Texas, December 9-11, 1980.

Pickard, Lowe and Garrick, Inc., "Seminar: Probabilistic Risk Assessment of Nuclear Power Plants," PLG-0154, November 1980.

Pickard, Lowe and Garrick, Inc., "Project Plan: Probabilistic Risk Assessment, Browns Ferry Nuclear Plant Unit 1," PLG-0149, October 1980.

Garrick, B. J., S. Kaplan, D. C. Iden, E. B. Cleveland, H. F. Perla, D. C. Bley, and D. W. Stillwell, "Power Plant Availability Engineering, Methods of Analysis - Program Planning - Applications," 2 Vols., PLG-0148, October 1980.

Bley, D. C., C. L. Cate, D. W. Stillwell, and B. J. Garrick, "Midland Plant Auxiliary Feedwater System Reliability Analysis," PLG-0147, October 1980.

Bley, D. C., D. M. Wheeler, C. L. Cate, D. W. Stillwell, and B. J. Garrick, "Reliability Analysis of Diablo Canyon Auxiliary Feedwater System," PLG-0140, September 1980.

Garrick, B. J., et al, "Project Plan: Oconee Probabilistic Risk Assessment," PLG-0138, August 1980.

Garrick, B. J., D. M. Wheeler, E. B. Cleveland, D. C. Bley, L. H. Reichers, and C. B. Morrison, "Operating Experience of Large U.S. Steam Turbine-Generators; Volume 1 - Data, Volume 2 - Utility Directory," PLG-0134, June 1980.

Garrick, B. J., S. Kaplan, and D. C. Bley, "Seminar: Power Plant Probabilistic Risk Assessment and Reliability," PLG-0127, May 1980.

Garrick, B. J., and S. Kaplan, "Oyster Creek Probabilistic Safety Analysis (OPSA)," presented at the ANS-ENS Topical Meeting on Thermal Reactor Safety, Knoxville, Tennessee, April 8-11, 1980.

Garrick, B. J., S. Kaplan, G. E. Apostolakis, D. C. Bley, and T. E. Potter, "Seminar: Probabilistic Risk Assessment as Applied to Nuclear Power Plants," PLG-0124, March 1980.

Garrick, B. J., S. Ahmed, and D. C. Bley, "A Methodology for Evaluating the Costs and Benefits of Power Plant Diagnostic Techniques," PLG-0118, January 1980.

Kaplan, S., B. J. Garrick, and D. C. Bley, "Notes on Risk, Probability, and Decision," PLG-0113, November 1979.

Bley, D. C., C. L. Cate, D. C. Iden, B. J. Garrick, and J. M. Hudson, "Seismic Safety Margins Research Program (Phase I), Project VII - Systems Analysis," PLG-0110, September 1979.

Cate, C. L., and B. J. Garrick, "W-501 Combustion Turbine Starting Reliability Analysis," PLG-0103, June 1979.

Pickard, Lowe and Garrick, Inc., "Plant Availability Program Specification, San Onofre Nuclear Generating Station," March 1979.

Pickard, Lowe and Garrick, Inc., "Work Order Scheduling System, Design Specification," March 1979.

D.F. Paddleford

Kansas State University, B.S. (1960) and M.S. (1962) in Nuclear Engineering

Post Graduate Courses in Engineering at UCLA, SMU, MIT

Mr. Paddleford is an Advisory Engineer in the Risk Assessment Section of the Nuclear Safety Department at the Westinghouse Nuclear Technology Division. Since joining Westinghouse PWR Systems Division in 1965, Mr. Paddleford has held a variety of positions in areas of increasing responsibility related to PWR plant safety, licensing, reliability, safety standards development, transient analysis, including core melt behavior, and probabilistic risk assessment. Most recently he has been engaged in the management and analysis of degraded core related issues, including test programs. He is currently active on AIF and IEEE Committees on Development of Risk Criteria and Utilization of PRA methods and is one of the principle authors of the Technical Writing Group which prepared the Industry/NRC "PRA Procedures Guides" under sponsorship of ANS and IEEE. He is a member of the IDCOR Technical Advisory Group and several IDCOR Expert Review Groups.

In the early 1970's his experience and responsibility included lead on research projects to develop a probabilistic approach to safety analysis, including systems reliability and data, probabilistic fracture mechanics models, core migration assessment, and probabilistic modeling of consequences associated with major fission product releases. Additional pertinent experience has included development of methods for parameter uncertainty propagation through design analysis computer codes and analysis of TMI and alternative scenarios at the request of the Kemeny Commission. Prior to joining Westinghouse, Mr. Paddleford was at Atomics International where he worked in areas of reactor physics and transient analysis in support of the SNAP 2/10 safety development.

Mr. Paddleford is a member of ANS and Sigma Xi and is a registered Professional Engineer. He is author or co-author of a number of papers on reactor safety and risk assessment.

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.

Garrick, B. J., S. Kaplan, G. Apostolakis, D. C. Iden, K. Woodard, and T. E. Potter, "Seminar: Probabilistic Risk Assessment of Nuclear Power Plants," PLG-0141, July 1980.

Garrick, B. J., S. Kaplan, G. E. Apostolakis, D. C. Bley, and T. E. Potter, "Seminar: Probabilistic Risk Assessment as Applied to Nuclear Power Plants," PLG-0124, March 1980.

Woodard, K., and T. E. Potter, "Modification of the Reactor Safety Study Consequences Computer Program (CRAC) to Include Plume Trajectories," presented to the 1979 ANS 25th Winter Meeting, San Francisco, California, November, 11-15, 1979.

Woodard, K., and T. E. Potter, "Assessment of Noble Gas Releases from the Three Mile, Island Unit 2 Accident," presented to the 1979 ANS 25th Winter Meeting, San Francisco, California, November, 11-15, 1979.

Garrick, B. J., S. Kaplan, P. P. Bieniarz, K. Woodard, D. C. Iden, H. F. Perla, W. Dicter, C. L. Cate, T. E. Potter, R. J. Duphily, T. R. Robbins, D. C. Bley, and S. Ahmed, "OPSA, Oyster Creek Probabilistic Safety Analysis," (Executive Summary, Main Report, Appendixes), PLG-0100 DRAFT, August 1979.

Woodard, K., and T. E. Potter, "Probabilistic Prediction of X/Q for Routine Intermittant Gaseous Releases," Transactions of the American Nuclear Society, Vol. 26, June 1977.

Dennis C. Richardson - Risk Assessment Technology Manager

Penn State University, B.S. Aerospace Engineering
1963

M.S. Control Engineering
1965

San Diego State University, M.S. Mathematics
1970

University of Pittsburgh, MBA
1980

Mr. Richardson has many years of professional and management experience in the nuclear field. He joined the Pressurized Water Reactor Division of Westinghouse in 1972 where he managed the Reactor Protection Analysis Group for performing nuclear plant safety analysis and, most recently, has managed the Risk Assessment Technology Organization.

Prior to this, Mr. Richardson was with Gulf General Atomic where he worked on design of control and safety systems for the gas-cooled nuclear plant. At Westinghouse, he has participated in and directed a number of risk assessment and safety analysis studies for a wide variety of applications. He was a principal investigator in both the Zion Station and Indian Point Station Reactor Safety Studies. He directed the PRA studies for the Westinghouse Owners Group that addressed the Post-TMI NUREG requirements on emergency procedures and operator display requirements. Mr. Richardson was technical and program manager for the British (NNC) Reference Water Reactor Safety Study. He has also led the development of economic and financial risk assessment techniques for the use in new reactor model design concepts.

Mr. Richardson is a member of the IEEE and ANS and has served on the working groups for two standards committees. He is reviewing the sections for the PRA manual directed by NRC to be finished in 1981. He is author or co-author of more than 15 reports and papers dealing with risk assessment and various aspects of nuclear plant design

* * * * * *

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

DOCKETED
USNRC

ATOMIC SAFETY AND LICENSING BOARD

'83 MAR 24 A10:55

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

OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

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

CERTIFICATE OF SERVICE

I hereby certify that on the 22nd day of March, 1983, I caused a copy of (1) Licensees' Testimony of Dennis C. Bley, Donald F. Paddleford, Thomas E. Potter, and Dennis C. Richardson on Commission Question Five, and (2) Licensees' Testimony of Thomas E. Potter on Commission Question Five, to be served by first class mail, postage prepaid on the following:

James P. Gleason, Chairman
Administrative Judge
Atomic Safety and Licensing Board
513 Gilmoure Drive
Silver Spring, Maryland 20901

Mr. Frederick J. Shon
Administrative Judge
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory
Commission
Washington, D.C. 20555

Dr. Oscar H. Paris
Administrative Judge
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory
Commission
Washington, D.C. 20555

Docketing and Service Branch
Office of the Secretary
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Joan Holt, Project Director
Indian Point Project
New York Public Interest Research
Group
9 Murray Street
New York, New York 10007

Jeffrey M. Blum, Esq.
New York University Law School
423 Vanderbilt Hall
40 Washington Square South
New York, New York 10012

Charles J. Maikish, Esq.
Litigation Division
The Port Authority of New York
and New Jersey
One World Trade Center
New York, New York 10048

Charles M. Pratt, Esq.
Stephen L. Baum, Esq.
Power Authority of the
State of New York
10 Columbus Circle
New York, New York 10019

Janice Moore, Esq.
Counsel for NRC Staff
Office of the Executive
Legal Director
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Brent L. Brandenburg, Esq.
Assistant General Counsel
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

Ellyn R. Weiss, Esq.
William S. Jordan, III, Esq.
Harmon and Weiss
1725 I Street, N.W., Suite 506
Washington, D.C. 20006

Charles A. Scheiner, Co-Chairperson
Westchester People's Action
Coalition, Inc.
P.O. Box 488
White Plains, New York 10602

Alan Latman, Esq.
44 Sunset Drive
Croton-On-Hudson, New York 10520

Ezra I. Bialik, Esq.
Steve Leipzig, Esq.
Environmental Protection Bureau
New York State Attorney
General's Office
Two World Trade Center
New York, New York 10047

Alfred B. Del Bello
Westchester County Executive
Westchester County
148 Martine Avenue
White Plains, New York 10601

Andrew S. Roffe, Esq.
New York State Assembly
Albany, New York 12248

Marc L. Parris, Esq.
Eric Thorsen, Esq.
County Attorney
County of Rockland
11 New Hempstead Road
New City, New York 10956

Phyllis Rodriguez, Spokesperson
Parents Concerned About Indian Point
P.O. Box 125
Croton-on-Hudson, New York 10520

Renee Schwartz, Esq.
Paul Chessin, Esq.
Laurens R. Schwartz, Esq.
Margaret Oppel, Esq.
Botein, Hays, Sklar and Hertzberg
200 Park Avenue
New York, New York 10166

Honorable Ruth W. Messinger
Member of the Council of the City of New York
District #4
City Hall
New York, New York 10007

Greater New York Council on Energy
c/o Dean R. Corren, Director
New York University
26 Stuyvesant Street
New York, New York 10003

Joan Miles
Indian Point Coordinator
New York City Audubon Society
71 West 23rd Street, Suite 1828
New York, New York 10010

Richard M. Hartzman, Esq.
Lorna Salzman
Mid-Atlantic Representative
Friends of the Earth, Inc.
208 West 13th Street
New York, New York 10011

Stanley B. Klimberg, Esq.
General Counsel
New York State Energy Office
2 Rockefeller State Plaza
Albany, New York 12223

Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Atomic Safety and Licensing Appeal Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Honorable Richard L. Brodsky
Member of the County Legislature
Westchester County
County Office Building
White Plains, New York 10601

Zipporah S. Fleisher
West Branch Conservation Association
443 Buena Vista Road
New City, New York 10956

Mayor George V. Begany
Village of Buchanan
236 Tate Avenue
Buchanan, New York 10511

Judith Kessler, Coordinator
Rockland Citizens for Safe Energy
300 New Hemstead Road
New City, New York 10956

David H. Pikus, Esq.
Richard F. Czaja, Esq.
Shea & Gould
330 Madison Avenue
New York, New York 10017

Amanda Potterfield, Esq.
New York Public Interest Research Group, Inc.
9 Murray Street, 3rd Floor
New York, New York 10007

David R. Lewis, Esq.
Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Mr. Donald Davidoff
Director, Radiological Emergency
  Preparedness Group
Empire State Plaza
Tower Building, Rm. 1750
Albany, New York 12237

Craig Kaplan, Esq.
National Emergency Civil
  Liberties Committee
175 Fifth Avenue, Suite 712
New York, New York 10010

Michael D. Diederich, Jr., Esq.
Fitgerald, Lynch & Diederich
24 Central Drive
Stony Point, New York 10980

Steven C. Sholly
Union of Concerned Scientists
1346 Connecticut Avenue, N.W.
Suite 1101
Washington, D.C. 20036

Spence W. Perry
Office of General Counsel
Federal Emergency Management Agency
500 C Street, S.W.
Washington, D.C. 20472

Stewart M. Glass
Regional Counsel
Room 1349
Federal Emergency Management Agency
26 Federal Plaza
New York, New York 10278

Melvin Goldberg
Staff Attorney
New York Public Interest
  Research Group
9 Murray Street
New York, New York 10007

Jonathan L. Levine, Esq.
P. O. Box 280
New City, New York 10958

Paul F. Colarulli