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

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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) : July 11, 1983

CON EDISON'S PROPOSED FINDINGS OF FACT REGARDING COMMISSION QUESTION 1

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COMMISSION QUESTION 1

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I. INTRODUCTION

1.0-1 Pursuant to the Commission's orders of January 8 and September 18, 1981, the Board ultimately admitted one contention as an issue in controversy under Commission Question 1 in this proceeding. In addition, the parties were invited to submit testimony concerning four Board questions. [Memorandum and Order (formulating Final Contentions and Setting Schedule), dated November 15, 1982].

1.0-2 The findings of fact below will first treat Commission Question 1, Board Question 1.1 and Contention 1.1 together, and will then address the remaining Board Questions seriatim.

II. COMMISSION QUESTION 1, BOARD QUESTION 1.1 AND CONTENTION 1.1

1.1-1 Commission Question 1 reads as follows:

> What risk may be posed by serious accidents at Indian Point 2 and 3, including accidents not considered in the plants' design basis, pending and after any improvements described in (2) and (4) below? Although not requiring the preparation of an Environmental Impact Statement, the Commission intends that the review with respect to this question can be conducted consistent with the guidance provided the Staff in the Statement of Interim Policy on "Nuclear Power Plant Accident Considerations under the National Environmen-

tal Policy Act of 1969;" 44 F.R. 40101 (June 13, 1980). 5/

1.1-2 UCS/NYPIRG proffered contention 1.1, which reads as follows:

> The probabilities and consequences of accidents at Indian Point Units 2 and 3 combine to produce high risks of health and property damage not only within the plume exposure EPZ but also beyond the plume exposure EPZ as far as the New York City metropolitan area.

1.1-3 Board Question 1.1 reads as follows:

> What are the consequences of serious accidents at Indian Point and what is the probability of occurrence of such accidents? In answering this question the parties shall address at least the following documents: (a) the Indian Point Probabilistic Safety Study (IPPSS) prepared by the Licensees; (b) the

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5/ In particular, that policy statement indicates that:

> Attention shall be given both to the probability of occurrences of releases and to the environmental consequences of such releases;
>
> The reviews "shall include a reasoned consideration of the environmental risks (impacts) attributable to accidents at the particular facility or facilities. . .";
>
> "Approximately equal attention should be given to the probability of occurrence of releases and to the probability of occurrence of the environmental consequences. . ."; and
>
> Such studies "will take into account significant site and plant-specific features. . ."
>
> Thus, a description of a release scenario must include a discussion of the probability of such a release for the specific Indian Point plants.

Sandia Laboratory "Letter Report on Review and Evaluation of the Indian Point Probabilistic Safety Study" (Letter Report) dated August 25, 1982; and (c) any other reviews or studies of the IPPSS prepared by or for the Licensees, the NRC Staff, or the Intervenors, or any other document which address the accuracy of the IPPSS.

A. SCOPE OF RISK ANALYSIS FOR THIS CASE

1.1-4 The NRC Staff and the Licensees, both of which have performed extensive risk analyses for Indian Point Units 2 and 3, have defined the phrase "serious accidents not considered in the plants' design basis" appearing in Commission Question 1 to be synonymous with severe core damage and core melt accidents. (Rowsome/Blond I, p. 4; Licensees 1, p. 9) No distinction is made for purposes of the Staff and Licensees' analyses discussed in these findings between severe core damage and core melt events. Both are referred to as core melt events. (Israel, et al., p. 6; Rowsome, Tr. 8279, 8308)

1.1-5 While there are several places in nuclear plants where radioactive materials are stored in addition to the reactor core, such as the spent fuel pool and the radioactive waste handling and treatment facilities, previous studies such as the Reactor Safety Study (WASH-1400) have shown unambiguously that both the risk posed by accidents and potential hazards posed by these materials

should an accident occur are far lower than for core melt accidents. Thus, we may confine our attention to the spectrum of accidents involving severe core damage or meltdown. (Rowsome/Blond I, p. 4)

B. RISK IN GENERAL

1.1-6 Risk entails a consideration of both the likelihood (probability) and severity of outcomes (consequences) of accidents. (Rowsome/Blond I, p. 6; Rowsome/Blond IV.A, p. 2; Licensees 1, p. 7). Both of these elements must be considered in order to accurately portray the risk associated with operation of Indian Point. To presume that a release occurs without accounting for its likelihood is a misrepresentation of risk. (Rowsome/Blond IV.A, p. 3; Sholly, Tr. 12750, 12771)

1.1-7 Risk can be presented on both an individual and societal basis. Societal risk expresses the health effects on a population group, i.e., the likelihood that any given number of people will be affected by an accident at a nuclear power plant. Individual risk is the likelihood that any given person in the vicinity of the plant will be affected by that accident. (Licensees 1, pp. 10, 19; Rowsome/Blond I, p. 4) The individual and

societal risks of accidents are expressed in terms of distinct measures such as early fatalities, radiation illnesses, thyroid cancers, latent cancer fatalities, total population dose and offsite property damage. (Licensees 1, pp. 18-19; Rowsome/Blond I, p. 6; Rowsome/Blond IV.A, pp. 2-3)

1.1-8 Mathematically, societal risk can be displayed as a family of curves that depict the severity of the consequences versus the likelihood or frequency of an outcome of at least that severity or greater. (Rowsome/Blond I, p. 6; Licensees 1, p. 19) Individual risk is expressed in terms of point values such as once in X years. (Licensees 1, p. 19)

C. PROBABILISTIC RISK ASSESSMENT IN GENERAL

1.1-9 Probabilistic risk assessment (PRA) is an advanced technique for quantifying the risks associated with complex systems such as nuclear power plants. The purpose of PRA is to develop mathematical models which result in explicit numerical calculations of the probabilities and consequences (risks) of accidental events. (Licensees 1, p. 6; Rowsome/Blond I, p. 7). This process provides an analytic framework for conducting careful sys-

tems analysis and for dealing with uncertainty. (Buchbinder/ Kubicki Testimony, p. 4) It is the only method of nuclear reactor safety analysis that accounts for multiple component failures, systems interactions and human error in an integrated way. PRA also uniquely provides a comprehensive framework for all aspects of reactor accident susceptibility. (Rowsome/Blond I, p. 12) Although it has strengths and weaknesses, PRA is the best means available to evaluate the risk of nuclear power plants. (Rowsome,Blond, Tr. 7225-7226) UCS/NYPIRG witness Weatherwax agreed that PRAs are the best modeling-derived estimates of risk. (Weatherwax Testimony, p. 4)

1.1-10 While it is possible to contrive scenarios by which most of the radioactive material in a nuclear plant could be released to the atmosphere and transported to the most populous location where it could do the most harm, the chance of this realistically occurring must be factored into the analysis. This realistic analysis is the objective of PRA. (Rowsome/Blond IV.A, p. 3)

1.1-11 In recent years, PRA methodology has developed rapidly and is being applied with increasing frequency in the nuclear industry in both industry-

sponsored and NRC Staff-sponsored efforts. At least 26 PRAs have either been initiated or completed in this country and overseas since NRC's publication of the Reactor Safety Study. (Licensees 1, pp. 6-7) PRA methodology has also been applied by the NRC Staff in its study of a variety of operating reactors as part of the Reactor Safety Study Methodology Applications Program (RSSMAP) and the Interim Reliability Evaluation Program (IREP). (Rowsome/Blond I, pp. 15-17)

1.1-12 PRA techniques have also been applied to the study of complex systems in other industries and programs such as NASA's space program, commercial aircraft studies, studies of protective systems for chemical plants and the public health risk assessment of a major chemical industrial facility in Great Britain. (Licensees 1, p. 7)

1.1-13 FOE/Audubon witness Dr. Charles Perrow alleged that PRAs are inappropriate for systems that are what he calls complexly interactive and tightly coupled because PRAs emphasize single failures, or common mode failures that are anticipated and understandable, and virtually ignore the interaction of small failures in independent subsystems that can unexpectedly become linked together. (Perrow Testimony, p. 4) His position is that

the complexity of nuclear power plants results in such incidents and that, because the plant is tightly coupled, there is little if any time to react once a sequence begins, so that incidents will propagate, leading to serious accidents. (Perrow Testimony, p. 3; Perrow, Tr. 7867) Further, he testified that in tightly coupled systems, there is little chance for operator intervention and recovery (Tr. 7866) and that, in any case, there is a high probability that such intervention would not be quick or thorough enough to prevent a catastrophe. (Perrow Testimony, p. 7)

1.1-14 As Dr. Perrow conceded, it is precisely because not all systems interactions can be anticipated that the defense-in-depth approach has been employed in nuclear plant design. (Tr. 7857-7858) Although Dr. Perrow cited the Browns Ferry, Three Mile Island, Dresden and Fermi accidents as illustrative of the problems inherent in tightly coupled systems, he conceded that in each of his examples intervention and recovery took place prior to any offsite radiation release. (Perrow Testimony, p. 4; Tr. 7868)

1.1-15 Dr. Perrow, a sociologist, has no formal engineering training. (Perrow, Tr. 7846) Although criti-

cizing PRA for its allegedly inherent omissions, Dr. Perrow had never reviewed a PRA for an operating facility prior to submitting his pre-filed testimony. (Perrow, Tr. 7848) Further, prior to his deposition in this case, he had never reviewed any portion of the IPPSS. (Perrow, Tr. 7846) Although he suggested that human factors had not been considered in the IPPSS, (Tr. 7850), Dr. Perrow testified that he did not review the human factors section of the IPPSS. (Tr. 7849) Dr. Perrow did not know which scenarios were considered by the IPPSS authors in preparing the study. (Tr. 7861, 7863) Staff witness Rowsome found little "coherence" in Dr. Perrow's testimony. (Rowsome, Tr. 8815)

1.1-16 The Board finds that, by Dr. Perrow's own definition, nuclear plants are not tightly coupled. Substantial time has been available in the past for operators to react, intervene and recover from incidents at nuclear plants prior to any serious offsite consequences. For Indian Point specifically, the record indicates that, for the most probable release category having serious offsite consequences (2RW), operators will have at least 12 and more likely 24 hours to intervene and recover safety system function. (See Finding 1.1-149) Dr. Perrow's lack of engineering experience, his lack of familiarity with

either PRAs for operating reactors in general or the IPPSS in particular, and the inherently contradictory nature of his testimony lead the Board to conclude that his testimony is of no probative value.

D. THE INDIAN POINT PROBABILISTIC SAFETY STUDY (IPPSS)

1.1-17 The IPPSS, which formed an important basis for the Licensees' testimony in this proceeding, is a massive and pioneering safety analysis for which the Licensees deserve a great deal of credit. It is a more comprehensive reactor risk assessment than any published in the United States heretofore. (Rowsome/Blond I, p. 25) Its preparation and publication required approximately two calendar years and 50 man years of intensive effort. (Licensees 1, p. 10; Rowsome/Blond I, p. 21) IPPSS was submitted to the Commission in March, 1982. (Licensees 1, p. 10) An amendment to the study was subsequently submitted to the NRC in January, 1983 (Bley, Tr. 13009), and its results were part of the Licensees' testimony. (Bley, Tr. 7031) The significant risk related items covered in the amendment were: (a) insertion of seismic bumper pads between the Unit 2 control building and the Unit 1 superheater building to reduce the seismic fragility of the Unit 2 structure and (b) provision of

alternate cables and transfer switches to assure that power is available to various critical components in the event of fire. (Bley, Tr. 7084-7085; IPPSS Amendment 1, 7.3-47A-1, 7.2-14A-1)

1.1-18 The IPPSS incorporates many advances over prior PRAs resulting from the continuing development of PRA techniques by the authors, the contributions of other consultants and government agencies in this field, the thoughtful critiques of previous PRAs by scientific peer groups, and the increasing body of nuclear power plant operating experience. (Licensees 1, p. 12) The study is noteworthy for its more thorough accounting of accident initiating events than prior PRAs, both those originating in the plant and those due to external events such as earthquakes and storms. The IPPSS has also broken new ground in the thoroughness of its investigation of challenges to the containment from severe reactor accidents. The IPPSS authors have also pioneered a technique for the propagation of uncertainties that can accommodate the effects of modeling approximations, phenomenological uncertainties and completeness as well as the more commonly treated statistical uncertainties. (Rowsome/Blond I, pp. 21, 25, 26; Licensees 1, pp. 12-18)

1.1-19 Even though it is difficult to attribute reliability to the very small probabilities reported in some PRAs (Rowsome, Tr. 7240-7241), the IPPSS has made significant progress in improving the understanding of and confidence in calculations of low probabilities. (Rowsome, Blond, Tr. 7174-7176)

1.1-20 Although several consultants, including Pickard, Lowe and Garrick, Inc., Westinghouse Electric Corporation and Fauske and Associates, Inc., each authored specific aspects of IPPSS, the analysis was an integrated effort among the consultants. (Garrick Tr. 7058-7060, Richardson, Tr. 7310-7311, Bley, Tr. 7311) The Licensees played an integral role in the IPPSS analysis to insure that the appropriate plant and site specific data were included. (Licensees 1, p. 11)

1.1-21 The IPPSS has received extensive review both by governmental experts and those in private industry. An independent review panel composed of prominent nuclear scientists provided critical peer review of the study. Sandia National Laboratory also coordinated a review of the IPPSS under contract to the NRC Staff. (Licensees 1, p. 11; Rowsome/Blond I, p. 25; NUREG/CR-2934) The NRC Staff reviewed both the IPPSS and

Sandia's analysis (Rowsome/Blond I, p. 25; Licensees 1, pp. 10-11) The Sandia and NRC reviews of the IPPSS consumed 5-10 man years of effort. (Rowsome, Tr. 7229) Although UCS/NYPIRG witness Weatherwax expressed the view that the Staff and Sandia reviews were not extensive enough to confirm the accuracy of the results (Weatherwax Testimony, pp. 19-20; Tr. 7922-7923), he spent only 75 to 80 hours reviewing the IPPSS. (Weatherwax, Tr. 7930)

1.1-22 The NRC Staff's review of the IPPSS culminated in the conclusion that the IPPSS employs approximations of risk that, on balance, are no less conservative than those employed in PRAs done by or for the Staff. (Rowsome/Blond I, p. 25)

E. METHODOLOGY

1.1-23 The four critical elements of any nuclear plant probabilistic risk assessment are (a) identification of initiating events, (b) plant response analysis, (c) containment response analysis, and (d) consequence analysis. (Licensees 1, p. 11; Rowsome/Blond I, pp. 7-11)

1.1-24 The first step in preparing a probabilistic assessment is the identification of initiating events,

i.e., disturbances in reactor operation that have the potential to lead to releases of radiation were it not for the intervention or presence of safety systems. (Rowsome/Blond I, p. 7) Such events require a rapid reactor shutdown and subsequent safety system operation to prevent core melt. (Israel, et al., p. 8) "Internal" initiating events are those which originate with malfunctions or failures of plant equipment or systems, including those caused by human error. "External" initiating events are those that originate from other causes, including earthquakes, fires, floods and high winds. (Licensees 1, p. 8)

1.1-25 "Event trees" and "fault trees" are used in the quantification of initiating events, plant response and containment response. (Licensees 1, pp. 8-9; Rowsome/Blond I, pp. 7-8) An event tree is a diagram that depicts the possible sequences of system functions and operator actions that could result from a given initiating event, including those which could possibly lead to core damage or to containment failure. The branches in the event trees represent the success or failure of the various safety systems or operator action in the plant. (Licensees 1, p. 8; Rowsome/Blond I, pp. 7-8) Fault trees are models of a system's reliability. They trace the origin of system failures to component failures and human errors. (Licensees 1, p. 9; Rowsome/Blond I, p. 8)

1.1-26 The catalogs of initiating events, event trees and fault trees are assembled into a mathematical model that permits accident sequence likelihoods to be calculated from initiating event frequencies, component failure rates, and human action and error rates. Rowsome/Blond I, p. 8) This plant response model delineates the functions needed to prevent core damage and to protect the containment. (Licensees 1, p. 52)

1.1-27 From the plant response analysis, groups of accident sequences with similar effects on the plant are combined into a reduced number of "plant damage states", and their total frequencies are determined. These physical conditions which are the results of the plants' response analysis are then the entry states (initial conditions) to the containment response analysis. (Licensees 1, pp. 63, 72) The containment event trees are quantified for each postulated accident sequence associated with that event tree. (Id., p. 80) The end point of these calculations is an evaluation of whether or not the containment building fails, how it fails, when it fails and what quantity of radioactivity and energy escaped from it to the biosphere. (Rowsome/Blond I, pp. 9-10)

1.1-28 The consequence analysis is the assessment of public health and economic impacts from postulated releases of radioactive material. The analysis estimates, using both deterministic and probabilistic input, the probability of various health effects from exposure to the releases assigned to each release category, given the occurrence of the release. (Licensees 1, p. 91; Rowsome/Blond I, p. 10) The results of the consequence analysis are used, along with the release frequencies determined in the plant and containment response analyses, to construct final risk curves. (Licensees 1, p. 91)

1. Family of Curves v. Point Estimates

1.1-29 In developing its estimates of risk, the Staff employed point estimates which were used to arrive at single-risk curves called Complementary Cumulative Distribution Functions (CCDF). (Acharya, p. III.C.A-39,n*) The authors of the IPPSS employed families of curves rather than point estimates to account for uncertainties associated with the risk estimates. These families of curves better convey the risk assessors's true state of knowledge. (IPPSS, p. 0-24; Licensees 1, p. 15; Blond, Tr. 8627, 8682) The preference in discussing risks is always to utilize families of curves to place risks in perspec-

tive and adequately address uncertainty. (Carrick, Tr. 6974) The family of curves represents the degree of confidence in the frequency numbers. (Kaplan, Tr. 6976)

1.1-30 Although the Staff did not use the family of curves approach, the Staff testified that the curves are more realistic than the point estimates because they account for uncertainties. (Blond, Tr. 8684) When performed accurately, the family of curves approach to risk used by the Licensees is the ultimate description of risk. (Blond, Tr. 8627) If the Staff analysis had been done with the family of curves approach, the Staff would have been able to more precisely indicate what level of uncertainty there is in their analysis. (Blond, Tr. 8812)

2. Statistical Techniques

1.1-31 The authors of the IPPSS employed a Bayesian method to update and integrate nuclear plant data. The use of Bayes' theorem is appropriate in such an analysis because it is a mathematical formula which allows quantification and integration of three important types of information: (1) general engineering knowledge and judgment; (2) industry-wide data; and (3) plant-specific data. (IPPSS, p. 0-91; Kaplan, Tr. 7052)

1.1-32 The chief controversy over the use of Bayesian methodology concerns the use and choices of prior distributions. (Easterling, Tr. 7400) A prior distribution expresses the state of knowledge prior to observing data. (Easterling, Tr. 7413) Thus, the use of prior distributions involves the expression of a state of knowledge in a quantitative probability distribution. Staff witness Easterling testified that this is difficult to do convincingly, and could mask uncertainty in the analysis. (Tr. 7400, 7403) As an alternative, Dr. Easterling recommended a different analytical method, the maximus method, a non-Bayesian approach which relies only upon "actual data". (Israel, et al., p. 13; Easterling, Tr. 7406, 7424, 7426-7427) The approach is basically to test the effect of the priors arrived at in the IPPSS by measuring the plant-specific data by themselves against the Bayesian posterior distribution resulting from a combination of the priors and the plant-specific data. (Easterling, Tr. 7424-7425)

1.1-33 Sandia National Laboratory recalculated important sequences from the IPPSS using the maximus methodology. These reports produced results consistent with those in the IPPSS. (Kaplan, Tr. 7322; Easterling, Tr. 7429) The differences between the Sandia-calculated core

melt frequencies and those in the IPPSS were not substantial, differing by only a factor of two or three. The changes that did occur as a result of Sandia' recalculations of core melt frequencies using the maximus approach were not due necessarily to statistical techniques, but more to systems analyses effects and to the component cooling water pipe break. (Israel, Tr. 7460; see Finding 1.1-77) Dr. Israel testified that it does not seem unreasonable for two different organizations to arrive at differences of that character. (Tr. 7477) Thus, by and large, the difference in statistical techniques thus had no effect. (Hickman, Tr. 7459, 7460) Staff witness Rowsome concluded that in view of the small difference in results produced using the Bayesian versus the maximus approach, the controversy over a proper statistical technique was not an issue affecting the validity of the risk calculations. (Tr. 8797)

1.1-34 The Bayesian methodology in the IPPSS produced reasonable point estimate failure probabilities. (Israel, et al., p. 10; Kolb, Hickman, Tr. 7459-7460) The mathematical approach used in the IPPSS was subjected to critical review by Sandia, and based on that review, the overall results did not change appreciably. (Israel, Tr. 7460)

1.1-35 While FOE/Audubon witness Dr. Isaac Levi did not object to the use of Bayes' theorem (Levi, Tr. 7765, 7787-7788), he questioned the choices of the prior distributions in its application. Specifically, he objected to the adoption of a definite prior distribution, and recommended that a much broader spectrum of such distributions be used. (Levi Testimony, p. 26, 28; Levi, Tr. 7729, 7771) According to Dr. Levi, the use of multiple priors would result in many "permissible" probability curves. (Levi Testimony, pp. 10, 25)

1.1-36 Because Dr. Levi believed that the indeterminacy (uncertainty) bands in the prior distributions used in the IPPSS were too narrow, he reassessed certain data from the IPPSS using a "K-factor", which adjusts the data by accounting for various levels of indeterminacy in the prior distribution. Dr. Levi characterizes the K-factor as an index of the extent to which one intends to be cautious in the initial assessment of prior probability. This factor also determines the worst permissible prior. (Levi Testimony, pp. 37-41; Levi, Tr. 7764-7765) In performing his reassessment of component failure rates, Dr. Levi chose factors ranging up to 99 because he believes that such high factors are necessary when

controversy exists concerning the degree of caution to be exercised. (Levi Testimony, pp. 40-41) However, Dr. Levi's calculations show that even when "K-factors" as high as 99 are employed to develop a worst case prior for a typical failure rate used in the IPPSS, the failure rate only differs by a factor of about ten in the median and a factor of about two in the mean. (Levi Testimony, p. 40; Levi, Tr. 7827)

1.1-37 Dr. Levi also objected to the frequent use of a lognormal distribution as a prior. (Levi Testimony, p. 34; Levi, Tr. 7740) In fact, the IPPSS did not use a strict lognormal distribution, but rather used discrete probability distributions, which often have certain characteristics of lognormal distributions. (Kaplan, Tr. 7108) Lognormal distributions are used extensively in risk and reliability work and closely conform to physical processes when the underlying variable is restricted to positive values. (IPPSS, p. 0-67)

1.1-38 Dr. Levi also asserted that the assessments of probability in the IPPSS were not sufficiently definite to justify a judgment in expected utility terms concerning the shutdowns of Indian Point. (Levy Testimony, p. 14) He proposed the use of the "minimax" theory, which

contrasts the worst possible consequences of operating Indian Point with the worst possible consequences of various alternatives. Probabilities would thus be ignored entirely. (Id., pp. 7-14) However, no analyses exist of the worst possible consequences of the various alternatives to operating Indian Point. Because individuals have their own criteria regarding worst possible consequences, the minimax theory may also not lead to a decision.

1.1-39 The minimax theory as proposed by Dr. Levi has never been employed in public policy decision making. (Levi, Tr. 7813-7814) Dr. Levi's approach is highly theoretical and academic in nature. (See Tr. 7713-7840) Dr. Levi conceded that there is substantial controversy surrounding the use of the minimax theory. (Levi Testimony, p. 9) Dr. Levi acknowledged that he is not an engineer and he is not familiar with the design details of the Indian Point plants. (Levi, Tr. 7790) Accordingly, Dr. Levi made his judgment that Indian Point should be shut down not as an expert, but in his role as an "ordinary citizen." (Levi, Tr. 7761)

1.1-40 Although Dr. Levi took it for granted that the Licensees based decisions in the IPPSS on the maximizing of expected utilities, he conceded that the

authors of the IPPSS did not present any results regarding expected utility. (Levi, Tr. 7817) In fact, the Licensees do not propose a method of making decisions based on maximizing expected utilities, but do recommend that explicit risk assessments be part of the decision-making process. (Carrick, Tr. 6984; Kaplan, Tr. 7256-7257, 7259)

3. Quantification and Propagation of Uncertainty

1.1-41 There are four kinds of uncertainties that affect PRAs. These are (1) statistical uncertainties, (2) modeling approximations, (3) errors of omission and (4) arithmetic errors in assembling the models. (Rowsome IV.C, p. 5)

1.1-42 The IPPSS authors pioneered a technique for the propagation of uncertainties which can accommodate the effects of modeling approximations, phenomenological uncertainties and completeness, as well as the more commonly treated statistical uncertainties. (Rowsome/Blond I, p. 26) The IPPSS is the most comprehensive quantitative assessment of uncertainties that has been presented in a PRA to date. (Rowsome IV.C, p. 4) While there is no generally accepted rigorous mathematical model for uncertainty analysis, (Weatherwax Testimony, p. 8; Easterling,

Tr. 7399), the approach used by the IPPSS authors to quantify uncertainty is on the forefront of the technology. (Blond, Tr. 8614) The IPPSS authors have identified, quantified, displayed and propagated uncertainties associated with their risk calculations throughout the IPPSS study. (Licensees 1, p. 15) Their treatment of uncertainty as an attempt to give their best judgment of the risks at Indian Point, was a reasonable way to proceed. (Rowsome, Tr. 8790)

1.1-43 The Staff as a whole made no formal attempt to quantify uncertainties in their own calculations of the risk posed by the Indian Point plants. (Rowsome IV.C, p. 2) A Staff attempt to use a different technique than that used in the IPPSS for quantifying uncertainty was not successful. Thus the IPPSS methodology is the only technique that has produced a measurable result. (Blond Tr. 8616-8617) Therefore, the Staff's approach throughout its analysis was to present a qualitative account of the sources of uncertainty in the calculation of risks asociated with Indian Point. (Rowsome IV.C, p. 3)

1.1-44 The uncertainties in the IPPSS appear large because uncertainties are proportionately greater for very small probabilities than they are for very large

probabilities. (Rowsome, Tr. 7174, 7240; see Kaplan, Tr. 7255)

(a) Statistical Uncertainty

1.1-45 Since the Licensees' use of Bayes theorem yielded results very similar to those derived by Sandia using the Maximus approach, the choice of statistical method is not a large source of uncertainty or potential error in most of the accident sequence likelihood estimates. (Rowsome IV.C, p. 6; see Finding 1.1-32 to 1.1-34)

1.1-46 The authors of the IPPSS used very broad priors which they believe overstated the uncertainty. This was an attempt to avoid the appearance of bias in the results. (Kaplan, Tr. 7107) For example, the IPPSS policy was to broaden, when appropriate, the distributions from the Reactor Safety Study (RSS) to extend the uncertainty band. (IPPSS, pp. 0-94 to 0-95) UCS/NYPIRG witness Weatherwax found this broadening procedure to be commendable. (Weatherwax Testimony, p. 17) In some cases, such as the interfacing systems LOCA analysis, the RSS distributions were judged to be sufficiently broad. (IPPSS, pp. 0-94 to 0-95) A re-analyis, using additional data not reported in the RSS, showed the frequency of this

accident sequence to be lower, not higher, by a factor of two to three than that reported in the IPPSS. (Easterling, Tr. 7456-7457) This analysis renders moot the issue of the breadth of the distributions for the interfacing systems LOCA. (Israel, Tr. 7423)

(b) Modeling Data Uncertainty

1.1-47 Modeling approximations, a second source of uncertainty in PRAs, are almost always taken in the pessimistic direction, i.e. they tend to exaggerate risk. An example is the treatment of partial failures. Safety functions that do not work as expected but do work partially are treated as outright failures. The risk assessment treats severe core damage, such as occurred at Three Mile Island, as a full core meltdown. Many practitioners of risk assessment believe that the exaggeration of risk predictions such as these is compensated for by errors of omission, although there is no reason to believe that this is always or precisely true. (Rowsome IV.C, p. 7)

1.1-48 A modeling approximation in the IPPSS which UCS/NYPIRG witness Weatherwax specifically criticized was the use of a constant exponentially distributed failure rate for plant equipment. (Weatherwax Testimony, p. 7; Tr. 7920)

1.1-49 On cross-examination, Mr. Weatherwax conceded that the IPPSS properly reflects the exponential failure rates that would exist in the plants with the data base that was used, but expresses the view that some pieces of equipment would behave otherwise, particularly with regard to the wear-out phenomenon. However, when questioned, Mr. Weatherwax was only able to point to one example of a piece of equipment which would behave otherwise than the norm. This was the reactor vessel, which would be subject to embrittlement leading to pressurized thermal shock events. (Weatherwax, Tr. 7957-7958) It should be noted that a PRA study prepared for Westinghouse plants demonstrated that pressurized thermal shock is not a contributor to risk. (Richardson, Tr. 7357)

1.1-50 The IPPSS did assign a constant exponential failure rate to plant equipment. However, since the failure rate data taken from the Indian Point plants includes both burn-in failures of new and newly replaced equipment as well as wear-out failures of older equipment, the plant equipment does have constant average failure rates determined from the variations in performance observed, given the continuing maintenance and replacement program at the plants. (Bley, Tr. 7381- 7382)

1.1-51 The specific concern raised by Mr. Weatherwax regarding pressurized thermal shock as a result of aging causing embrittlement was addressed by both the Licensees and the NRC Staff in their direct testimony on Board Question 1.4. The risk resulting from pressurized thermal shock events is very low for the Indian Point units. (See Findings 1.4-1 to 1.4-21)

1.1-52 A second specific area of modeling uncertainty which was raised concerns the issue of human reliability. The human error analysis in IPPSS represents a substantial extension of the human reliability analysis contained in the RSS. (IPPSS, p. 0-99) Despite some shortcomings, IPPSS human reliability analysis represents a detailed, thoughtful, and objective attempt to analyze the most difficult-to-analyze system component -- the human. [NUREG/CR-2934, "Review and Evaluation of the Indian Point Probabilistic Safety Study," p. 2.5-1 (Dec. 1982)] The authors of IPPSS used the Commission's Human Reliability Handbook as the principal source of human error rate information. (Licensees 1, p. 16) The data in the Handbook are "derived data," that is, they are based on either experimental data or data from other kinds of operations. (Swain, Tr. 7444-7445) The authors of IPPSS modified the

failure probabilities given in the Handbook to account for plant-specific situations, a practice recommended by the Handbook's author. (Swain, Tr. 7442)

1.1-53 Stressful or unusual situations are particularly difficult to model. (Swain, Tr. 7441) In the IPPSS, the human error rates for such events as fire were increased to reflect circumstances, such as relatively high stress and competing demands for the operators' attention. Senior Reactor Operators' availability and the benefits from operator training on a plant-specific simulator were also considered. These human error rate distributions were then used to quantify the plant and system logic models. (Licensees 1, p. 16)

1.1-54 While some areas of human reliability can be modeled well, e.g. errors of maintenance and surveillance, the difficult aspects of this subject to model are broad cognitive pattern recognition, and the creativity operators can use in solving problems. (Rowsome Tr. 7238) One specific area which is difficult to assess is the likelihood that operators might misdiagnose an accident and so employ the wrong procedures. (Rowsome/Blond I, p. 12; Rowsome, Tr. 7226)

1.1-55 Regarding situations in which several operators may make the same mistake, such as cognitive errors leading to misdiagnosis, IPPSS did not model the human error rates as independent probabilities. Rather, IPPSS analysts used the dependence model in the Handbook, which assumed varying degrees of dependence among the levels of operators based upon the activities in which the operators were expected to be involved. IPPSS analysts made judgments as to the appropriate levels of dependence. (Bley, Tr. 7367; Swain, 7447-7448) This was accomplished by increasing the error probabilities for each additional operator, given that the first operator had made an error. (See Swain, Tr. 7448-7449)

1.1-56 With increased operator training and extensive simulator training for operators whose performance relating to accident diagnoses is being modeled, the gap between PRA methodology and actual experience will converge. (Rowsome, Tr. 7226-7227) Such increased operator training has been implemented since the Three Mile Island accident, and simulator training is in place at both Indian Point Units. (Rowsome, Tr. 7227) As a result, TMI-type accidents involving operator misdiagnosis are now far less likely to occur. (Rowsome 1.2, p. 11) Indeed, the Sandia review conducted by Dr. Swain determined the

likelihood of such operator misdiagnosis to be negligible. (Swain, Tr. 7449-7450)

1.1-57 Operators can also have a positive influence on the course of accidents. (Israel, et al., p. 9) In general, the authors of IPPSS made a deliberate decision not to take full credit for certain operator recovery factors. (NUREG/CR-2934, pp. 2.5-13)

1.1-58 The human error probability estimates were reviewed in depth by the Staff for four important activities for which there were emergency procedures. The Staff concluded that these estimates in IPPSS were low in two activities, and high in the other two.. (Israel et al., pp. 9-10). Regarding the analysis of common cause failures from human error in IPPSS, the Staff found that, upon further checking, certain omissions in the information provided made no difference. (Swain, Tr. 7446)

(c) Errors of Omission

1.1.59 The authors of IPPSS should be commended for greater use of plant specific initiating event and component failure data than found in many past PRAs. (Israel, et al., p. 10) IPPSS data base also included recent

industry-wide nuclear plant operating experience data. After the industry-wide data were collected, plant records, drawings, and system descriptions were reviewed to construct the plant-specific data base. The industry-wide data were then integrated with the plant specific data using Bayes' theorem, thus permitting the use of all relevant information to reflect the current state of knowledge. IPPSS data base includes component failure rates, component unavailability due to testing and maintenance, human error rates, site seismicity, frequencies for initiating events such as wind, tornado, and fire, and meteorological, topographic, demographic, and site-specific evacuation site-data. (Licensees 1, pp. 13, 15)

1.1-60 The Staff's initial review of the IPPSS raised questions regarding the particular data used; subsequent communications with the authors of IPPSS revealed valid reasons for the choices made. (Kolb, Tr. 7417) For example, the Staff testified that it was appropriate not to rely on the Nuclear Plant Reliability Data System because questions exist about its completeness and accuracy. (Easterling, Tr. 7455)

1.1-61 Although the IPPSS systematically attempted to identify and quantify all conceivable accident sce-

narios that could significantly threaten the public, other scenarios are possible (Licensees 1, pp. 49-50), and it is conceivable that some cutsets (failure modes) may have been omitted from the analysis. (Rowsome, Tr. 8784, 8785) However, scenarios which might have been overlooked are likely to be rare because of the thoroughness of the analysis (Licensees 1, pp. 49-50; see Rowsome IV.C, p. 8), and because the Sandia and Staff reviews were comprehensive enough to find, recognize and correct most hypothetical errors or distortions. (Rowsome, Tr. 7217) Because of the care taken in modeling the more likely and more severe accidents, most errors of omission are likely to involve less important contributors to risk. (Rowsome/Blond IV.C, pp. 8-9)

1.1-62 Even if a previously unidentified accident were to occur, it would have to either bypass or breach containment to pose a risk to the public. Very few accidents are capable of doing this, as the IPPSS analysis demonstrated. (Licensees 1, pp. 50-51) The consequences of an unidentified scenario are likely to be no more, and probably less, severe than those already investigated. This is because, among the identified scenarios, infrequent weather conditions are necessary to obtain the maximum consequences. (Licensees 1, p. 51; see Rowsome/Blond IV.C, pp. 8-9)

1.1-63 Relying on their past PRA experience, analysts at Sandia National Laboratories examined the "building blocks" in IPPSS to determine if possible errors, unrealistic assumptions, or omissions made by IPPSS analysts could allow additional sequences to become important. With one exception, a pipe break in the component cooling water system, the initiating events covered in IPPSS were "relatively complete." (Israel, et al., pp. 7, 8) Ongoing analyses by the Licensees indicate that the probability of such a pipe break occurring is extremely remote. (Bley, Tr. 12699-12700) Thus, the contribution to the overall risk from unidentified scenarios is thought to be small. (Licensees 1, p. 51)

1.1-64 Specific completeness issues were raised by UCS/NYPIRG regarding the failure of IPPSS to treat sabotage, and the manner in which common cause failures were analyzed. (Weatherwax Testimony, pp. 4-6; Weatherwax, Tr. 7920)

1.1-65 Multiple failures of safety equipment can result from a common cause. Failures of common support equipment such as power supplies or cooling systems have been explicitly modeled in the plant event trees or system

fault trees. Human errors such as faulty maintenance, misinterpretation of plant conditions, and sequentially compounded errors were included in the systems' analyses. Initiators such as earthquakes or tornadoes that could disable multiple systems were explicitly modeled. Other environmental factors with common cause potential, such as excessive humidity or dirt, were considered in each system analysis. (Licensees 1, p. 59)

1.1-66 The Staff's analysts determined that IPPSS modeled the more important of these common cause failures. (Israel, et al., p. 10)

1.1-67 IPPSS has advanced PRA methodology in the areas of common cause failures through its treatment of such factors as human errors, including maintenance errors; the modeling of external events, such as earthquakes, that can produce multiple failures; and failures of common support equipment, such as power supplies or cooling systems. (Licensees 1, pp. 15-16, 59) The authors of IPPSS developed a cause table for each category of minimal cutsets, those minimal combinations of component failure which lead to system failure, (IPPSS, pp. 0-104, 0-107), a procedure endorsed by intervenors to preclude the omission of common mode failures. (Weatherwax Testimony, p. 12)

1.1-68 Although IPPSS modeled the more important common cause failures, since its publication new data sources concerning common cause failures have become available. Because these new data suggest that common cause failures may be more frequent than previously believed, the Staff included them in their analysis. (Israel, et al., pp. 10-11) Although no attempt was made by the Staff to determine the incremental effect of these new data on the actual public health risk, Staff witness Kolb cited other factors as being more important than these data in developing Staff's revised plant damage state probalities. (Kolb, Tr. 7457-7458)

1.1-69 An additional advance in the area of common cause failures is the use in IPPSS of the beta factor model, (Licensees 1, p. 16), where "beta" is the conditional probability of redundant subsystem failure, given that failure of a single redundant component occurs. (IPPSS, pp. 1.3-3 to 1.3-4) By reviewing data sources for "beta", i.e. common cause failure data, the IPPSS authors identified underlying causes. By detailed plant-specific quantification of the list of causes, the unavailability of plant systems due to common causes was determined. Utilizing this methodology, some residual contributions by

identified "other" causes remains but the magnitude of that contribution is greatly reduced. This approach provides a framework for using a wide range of experience to analyze rare common cause contributions. (IPPSS, 1.3-4) Although making a general criticism of the IPPSS treatment of common mode failures (Weatherwax Testimony, pp. 4-5), Mr. Weatherwax did not offer any specific criticism of the "beta factor" approach used in identifying common cause failure initiators. He was unable to identify any particular common mode failures, which had been omitted in the IPPSS and would have been surprised if he had identified any failures which Sandia had not already addressed. (Weatherwax, Tr. 8028)

1.1-70 Regarding sabotage, this subject matter was specifically omitted from the IPPSS. (Licensees 1, p. 49 n.1; Bley, Tr. 7031) It was also intentionally omitted from the Staff risk analyses. (Israel, et al., p. 5) Therefore, no finding can be made concerning its contribution to risk. (Bley, Tr. 7031; Garrick, Tr. 7039) Sabotage was excluded from the IPPSS because its treatment was considered not to be appropriate or compatible with publication in a publicly available document. (Garrick, Tr. 7043, 7050, 7147) A probabilistic assessment of sabotage would require a very detailed discussion of possible

sabotage scenarios and plant security arrangements to deal with those scenarios. Placing such information in the public domain would not be responsible. (Bley, Tr. 7150) The interests of nuclear power plant safety would not be well served by such a public discussion of this issue. (Garrick, Tr. 7050) Furthermore, the Staff testified that it was not aware of any method which accurately predicts the likelihood of sabotage attempts. (Rowsome/Blond I, p. 12)

1.1-71 The Board finds that sabotage is a very security-sensitive issue. The Licensees' and Staff's judgment was therefore justified that although it could be treated with PRA methodology, sabotage should not be treated in a plant-specific probabilistic risk assessment. (See Garrick, Tr. 7043, 7045-7046, 7145, 7147) There is nothing in the record to suggest that the Indian Point plants are any more susceptible to sabotage than any other plant. With the exception of sabotage, which was explicitly and reasonably omitted, no significant omissions from IPPSS have been identified. (Bley, Goeser, Garrick, Kaplan, Tr. 7030-7031)

1.1-72 Given the IPPSS technique of including the "other" accident category (which is the universe of

everything else that has not been considered by the analyst), there is not much more that can be done to assure completeness. (Blond, Tr. 8615, 8616)

(d) Arithmetic Uncertainty

1.1-73 In principle, arithmetic errors could grossly distort the results of a PRA. However, a significant distortion of the risk due to arithmetic errors in either the IPPSS or the Staff calculations of risk would have been conspicuous in the comparison of the two studies and against the background of other PRAs and risk research. Thus, it was unnecessary to count upon formal checking procedures to exclude the possibility that arithmetic errors are responsible for large distortions of the risk profile of the plant. (Rowsome/Blond IV.C, p. 12)

F. INITIATING EVENTS

1.1-74 Examination of the Indian Point operating reports, Electric Power Research Institute (EPRI) reports, and the Commission's Operating Units Status Reports was undertaken to obtain plant-specific and industry-wide data on the frequency distribution for initiating events. The latter data is derived from 30 nuclear plants. The plant-

specific frequency distributions that resulted were then used in quantifying the plant event tree sequences. (Licensees 1. p. 62)

1. Internal Initators

1.1-75 Internal initiating events include loss of feedwater flow, loss of AC or DC power, turbine trip, operator error, and pipe failures. (Licensees 1, pp. 41, 44) Imbedded in IPPSS initiating event data are human interactions, because they can directly cause initiating events. (Id., p. 61)

1.1-76 Internal initiating events were identified through the use of a number of sources, including the Final Safety Analysis Report (FSAR), Licensee Event Reports (LERs), EPRI reports, the RSS, various Westinghouse studies, and Indian Point plant-specific reports. (Licensees 1, pp. 41-42) The identified events were then placed into one of twelve initiating event categories in a core damage master logic tree. (Id., Figure IV-1, p. 43) These categories include large, medium and small break LOCAs, leakage to secondary coolant, partial loss of steam flow, turbine and reactor trips, steam releases inside and outside containment, and core power increase events. (Id., p. 44)

1.1-77 The initiating events included in IPPSS are relatively complete and their frequency estimates are reasonable compared to those addressed in previous PRAs. (Israel, et al., p. 8) The Staff did testify that there was no indication that the IPPSS had considered an initiating event caused by a pipe break in the component cooling water system. (Id.) However, continuing work on the component cooling water system by the IPPSS analysts has shown that the chances of a large break disabling the system are extremely remote due to the quality of the piping, the pressures involved, the make-up capabilities of the system, and the energy required to shear one of the major pipelines. (Bley, Tr. 12699, 12700)

1.1-78 UCS/NYPIRG concedes that the initiating event frequencies in IPPSS range from reasonable to quite conservatively high. (Weatherwax Testimony, pp. 9, 11)

2. External Initiators

1.1-79 External initiating events were compiled by the authors of IPPSS using the FSARs as a basis and updating them with information and data bases from federal and local government agencies, and from appropriate indus-

trial and commercial organizations. (Licensees 1, p. 42) Among the external initiating events considered in IPPSS were earthquakes, fires, turbine missiles, flooding from plant system and external sources, wind and wind induced missiles, and transportation and storage of hazardous materials. The most significant external initiating events are earthquakes, wind and fires. (Licensees 1, pp. 42, 45)

(a) Seismic

1.1-80 IPPSS analysis of earthquake risk has broken new ground in its methodology and use of advanced data. Compared to the RSS, IPPSS has achieved "major new engineering insights." (Budnitz Testimony, pp. 7, 13) The methodology used in IPPSS to analyze the risks from earthquakes, which is discussed in the PRA Procedures Guide, NUREG/CR-2300, involves four steps: (1) the seismic hazard analysis, which involves a determination of the likelihood of various sized earthquakes near the site and the level of ground motion they can cause at the site; (2) the fragility analysis, which assesses the ability of plant structures and components to function under seismic stress; (3) the plant analysis, which traces a seismically-induced accident through the plant systems; and (4) the consequence analysis, which determines the radiological

effects of a seismically-initiated accident. (IPPSS, p. 7.2-1; see Budnitz Testimony, pp. 12-13)

1.1-81 In the IPPSS as initially published, the core melt frequency for Indian Point Unit 2 was dominated by interaction between the Unit 1 superheater building and the Unit 2 control building. Since the IPPSS was initially published, after investigating various ways of reducing the core melt contribution from this event, Con Edison decided to change the physical configuration by widening the clearance between the two buildings at the roof line, and by inserting rubber pads in the gap between the two buildings to absorb impact energy in the unlikely event of an earthquake that would cause the accident scenario revealed in the IPPSS. (Budnitz Testimony, pp. 18, 19). Con Edison has re-examined the seismic contribution to overall mean core melt frequency for Unit 2, taking into account the modification and that frequency has been reduced from 1.4×10^{-4} to 7.9×10^{-6}. (IPPSS Amendment 1, p. 7.2-14A-1; Budnitz Testimony, p. 19)

1.1-82 Use of ground acceleration (g) is the appropriate method of assessing the seismic capacity of a structure. This involves determining the earthquake's magnitude or intensity and then calculating the attenua-

tion from the potential source point to the plant and the acceleration at the plant. Ground acceleration cannot be directly related with any accuracy to other earthquake scales, such as Modified Mercalli or Richter, because of distance and ground media attenuation effects between the source and the point of interest. (Budnitz, Tr. 7502; Perla, Tr. 13007) The only circumstance in which these scales would be directly comparable would be if the epicenter of an earthquake were directly underneath a nuclear plant. (Perla, Tr. 13007-13008)

1.1-83 A recent reanalysis of the seismic fragility of the Indian Point containments has shown that while a ground acceleration of more than 0.8g is not possible from any earthquake that could affect the Indian Point site, (Perla, Tr. 13007-13008), the Indian Point containments have a median structural capacity of 2.7g. [Perla, Tr. 13004; letter from D. Wesley to H. Perla and Attachments (Wesley Letter, Dec. 30, 1982), exhibit to Licensees 1, admitted Apr. 7, 1983, Tr. 12989] Therefore, direct failure of the containments as a result of earthquake is impossible. (Perla, Tr. 12992) As a result, the Z-1Q release category caused by a seismic event and described in the IPPSS is no longer applicable. (Bley, Perla, Tr. 7290-7291; see Budntiz, Tr. 7531)

1.1-84 Staff stated that the re-analysis of the Indian Point containment buildings, performed by competent analysts, confirms their judgment that the original IPPSS analysis was conservative, that is, it understated the strength of the containment. (Budnitz, Tr. 7528-7530) Although this analysis shows that the containment cannot directly fail from any seismic event which could occur in the area, Staff has retained a failure mode for this accident until a detailed review of the recent analysis is completed. (Meyer, Pratt, Tr. 12507, 12509) The overall risk would be reduced with elimination of this failure mode (Budnitz, Tr. 7531-7532; Meyer, Tr. 12509), since half of the early fatality risk for Indian Point had come from that release category. (Budnitz, Tr. 7531-7532)

1.1-85 The largest acceleration expected in the vicinity of Indian Point within a 500-year period is less than 0.1g. (Licensees 1, p. 46) There are almost no recorded earthquakes in the northeastern United States of a magnitude that could affect reactor risk. (Budnitz Testimony, p. 20) Earthquakes with ground accelerations of about 0.1g to 0.3g could result in injuries and loss of life to the public from heavy damage to conventional struc- tures such as residences and office buildings. However, such accelerations would not be great enough to

threaten even the weakest critical structure or equipment at the Indian Point plants. (Licensees 1, p. 47)

1.1-86 Staff maintained that it would be useful to obtain PRA-type insights into how the Ramapo fault zone might interact with the Indian Point plants, although specifically disclaiming any implication of significant concern that the Ramapo zone might be a significant risk contributor. (Budnitz Testimony, p. 24) IPPSS considered the Ramapo fault as part of larger source areas. (See IPPSS §7.9.2, p. 6; see also NUREG/CR-2934, Appendix A, p. 3-7) According to Sandia National Laboratories, strong evidence does not exist to support a hypothesis of a Ramapo source. (NUREC/CR-2934, pp. 2.7.1-3, 1-4)

1.1-87 A Ramapo source hypothesis would be assigned a very low probability and, therefore, more explicit consideration of this potential fault zone would not affect IPPSS results in a significant way. (NUREG/CR-2934, Appendix A, pp. 3-6 to 3-7) Dr. Budnitz suggested that the use in IPPSS of sustained acceleration probably gives too much weight to small, nearby, localized earthquakes in relation to their actual contribution to risk. (Budnitz Testimony, p. 25)

1-1-88 Peak ground accelerations at which failure occurs reflect the response of plant structures, equipment, and components to seismic events. In IPPSS, failure of structures housing vital equipment or components was defined to occur when structural deformations, which do not necessarily lead to collapse, interfere with the operability of equipment attached to the structure. Electrical and mechanical equipment failure was defined to occur when essential function capability was lost. Failure of piping was defined to occur when the pressure boundary ruptured. (Licensees 1, pp. 64-65)

1.1-89 The Indian Point Units 2 and 3 safety related structures and equipment are designed to withstand, without loss of function, the design basis earthquake of 0.15g. Detailed studies of these safety related structures and equipment show that their actual capability to withstand seismic events considerably exceeds the design basis value. (See IPPSS Tables 7.2-3, 7.2-7; Licensees 1, p. 65) The lowest median acceleration capacities for safety related equipment are in the range of 0.7g to 0.9g. Even the least capable of these components is likely to function during very large earthquakes. Safety related equipment with even higher median acceleration capacities can limit the consequences of seismically induced failures. (Licensees 1, p. 65)

1.1-90 Sandia National Laboratories was initially concerned regarding the seismic capacity of the control room ceiling. (NUREG/CR-2934, Appendix A, p. 5-2) That concern was explicitly addressed in Amendment 1 to IPPSS. (IPPSS Amendment 1, pp. 7.1-21 to 7.2-21A-2). The Amendment provides further documentation of the control room analysis already presented in IPPSS. (Perla, Tr. 13016-13017). In addition, modifications to the control room ceilings and reanalyses of their possible failure frequencies resulted in the conclusion that they no longer contribute significantly to seismic risk. (Wesley Letter, Dec. 30, 1982)

1.1-91 Significant conservatisms (e.g., overstatements of risk) exist in IPPSS seismic analysis, including the use of the lognormal distribution to model component capacities. This is because the tails of such distributions do not accurately reflect the behavior of actual physical systems. The tail of a lognormal distribution extends further than is actually the case for the distribution of the physical motion. (Budnitz Testimony, p. 24)

1.1-92 The Staff testified that the seismic uncertainties in IPPSS, although large, may be understated because there is greater uncertainty regarding the frequency of events that have never occurred than there is regarding those that have occurred. (Budnitz Testimony, p. 8; Budnitz, Tr. 7532) Although it is difficult to quantify the uncertainties in the overall risk numbers, or to affirm the validity of the central values in IPPSS, the seismic portion of the study includes overstatements which, if corrected, would lower the ultimate calculated risk. These overstatements exist because the authors of IPPSS made "pessimistic" rather than "realistic" assumptions in the seismic analysis. (Budnitz Testimony, pp. 30-33) A more thorough seismic analysis (which is beyond the Staff's present capability) would result in the conclusion that the Indian Point buildings are stronger than the present Staff analysis shows. (Budnitz, Tr. 7557)

1.1-93 Independent analyses of the estimates of seismic risk at Indian Point confirm that no significant differences exist between the results of such analyses and those in IPPSS. (Budnitz Testimony, pp. 35-36; Budnitz, Tr. 7512) Even with the large uncertainties involved, the IPPSS treatment of seismic events is not only the best such analysis that has ever been done, but is a true breakthrough in this area. (Budnitz, Tr. 7539)

(b) Fire

1.1-94 The evaluation of fires in the IPPSS is composed of six principal steps: identification of critical areas within the plant; calculation of the frequency of fires in these areas; calculation of the probability of fire propagation, considering fire barriers and extinguishing equipment; assessment of potential fires that might initiate an accident; assessment of the effect of fires on mitigating systems; and calculation of the frequencies of the various plant responses and release categories which could result. (Licensees 1, pp. 65-66) The determination of frequency of fires at the plant was based upon American Nuclear Insurers data, LER's, the technical literature, engineering judgments and conservative assumptions supplemented by historical evidence. (Licensees 1, pp. 47- 48)

1.1-95 Following a preliminary analysis of potential fires throughout the plant, the authors of IPPSS conducted a more detailed, on-site evaluation for those critical locations where a fire might damage important equipment or cables, and thus initiate an accident scenario. The authors of IPPSS believed that very detailed analyses

of all plant locations would not significantly change the results. (IPPSS, p. 0-126; Bley, Tr. 7030)

1.1-96 UCS/NYPIRG argued that more locations should have been analyzed. (Weatherwax Testimony, p. 15) In Amendment 1 to the IPPSS, the analysis was expanded to cover a great many more areas than were analyzed originally. No significant changes in risk resulted from this reanalysis. (IPPSS Amendment 1, p. 7.3-111A-1; Bley Tr. 7030)

1.1-97 According to Staff, control room fires should have been comprehensively analyzed in IPPSS because of the potentially large consequences which could result from such events. However, Staff stated that there are no data to support a high estimated frequency of serious control room fires. (Buchbinder/Kubicki Testimony, pp. 9-10) UCS/NYPIRG claimed that no consideration was given in the IPPSS to control room fires. (Weatherwax Testimony, p. 15) Licensees provided a more detailed analysis of potential fires in this area of the plant in Amendment 1 to IPPSS. Because this area is continuously manned, a fire would be detected almost immediately and attempts to extinguish it would begin promptly. The results of this analysis showed that the frequency of core melt accidents originating from control room fires accounts for only a

small fraction of the other fire-initiated core melts. (IPPSS Amendment 1, pp. 7.3-111A-15 to 7.3-111A-16)

1.1-98 Fires in critical locations are unlikely at Indian Point because of such precautions as the emphasis on nonflammable materials, administrative controls, fire barriers, and fire detection and suppression systems. Fires involving large quantities of combustible materials in critical locations are the only ones that could result in serious damage to the plant. Because such conditions do not exist at the plant, only large quantities of transient combustibles are of concern. (Licensees 1, p. 48)

1.1-99 The authors of IPPSS acknowledge that fire analysis involves complex issues which have not received as detailed attention as have other aspects of risk assessment in previous studies. For this reason, major assumptions in the IPPSS fire analysis were intentionally conservative. (IPPSS, p. 7.3-3) The lack of historical data is a major contributor to the wide uncertainty range, together with the complexity of the physical and chemical process phenomenology, which makes fire behavior difficult to assess. (Buchbinder, Tr. 7580-7581, 7585) While the authors of IPPSS may have been

overly optimistic in their assessment of safe shutdown through the use of local auxiliary feedwater pump controls given a fire in the cable spreading room, overestimates were introduced into IPPSS analysis in an attempt to compensate for the uncertainties in the fire risk modeling. (IPPSS, p. 7.3 -3; see Buchbinder/Kubicki Testimony, pp. 11-12) These included conservatisms in fire growth and spread modeling leading to overestimates of cable ignition probabilities and conservatism in modeling fire extinguishment effectiveness of sprinklers in electrical tunnels. (Buchbinder/Kubicki Testimony, pp. 12-13; Buchbinder, Tr. 7584)

1.1-100 Staff stated that the assumptions used to model fire growth and spread are "extremely conservative." (Buchbinder/Kubicki Testimony, pp. 12-14; Buchbinder, Tr. 7584) The Staff analysis did not include a correction for this overly cautious analysis in IPPSS. (Buchbinder/Kubicki Testimony, p. 14)

1.1-101 Staff asserted that an increase in the Indian Point Units 2 and 3 fire-initiated core melt frequency of factors of two and of four respectively might be appropriate if high temperature degradation of cable insulation is considered. (NUREG/CR-2934, p. 5.1-6; Buchbind-

er/Kubicki Testimony, P. 10; see Weatherwax Testimony, p. 15) However, Staff acknowledged that its estimate should not be considered accurate, but should be considered as a sensitivity analysis. (Buchbinder/Kubicki Testimony, p. 10)

1.1-102 Staff stated that fires in non-critical areas that could spread to critical areas should have been analyzed. (Buchbinder/Kubicki Testimony, p. 10) UCS/NYPIRG concurred. (Weatherwax Testimony, p.15) However, Staff asserted that there are no fire loss data adequate to demonstrate that such fires do indeed make a significant contribution to risk. (Buchbinder/Kubicki Testimony, pp. 10-11)

1.1-103 Appendix R to 10 CFR Part 50, which became effective on February 17, 1981, provides new licensing requirements to reduce fire risk. Implementation and enforcement of Appendix R is in progress for all nuclear power plants, including Indian Point Units 2 and 3. (Buchbinder/Kubicki Testimony, pp. 5, 19-25) Modifications at the plants include the capability to supply power to various safety components by routing additional cables which are physically separated from normal cable routes. Because of this additional cable routing, fires

in locations such as the cable spreading room, the switchgear room, the electrical tunnel and penetration area, and the motor control center area are far less likely to have any risk implications. (Licensees 1, pp. 66)

1.1-104 The Staff's testimony conveyed the impression that it had identified these modifications to Con Edison as steps which should be taken to reduce fire risk at Indian Point 2. (Buchbinder/Kubicki Testimony, p. 20) However, on cross examination, the Staff conceded that it had merely confirmed plant modifications to reduce fire risk which Con Edison had previously decided to implement. (Kubicki, Tr. 7590) These modifications to Unit 2 yield a reduction in mean core melt frequencies for five sequences of approximately one order of magnitude for that facility. (Buchbinder/Kubicki Testimony, pp. 22, 23)

1.1-105 After the backfits at both units to achieve compliance with Appendix R, the risk of fire will be even further reduced, and fire will not be a major contributor to core melt frequency. (Buchbinder/Kubicki Testimony, pp. 5, 6)

1.1-106 The authors of IPPSS are to be "congratulated" for their effort in the area of fire analysis.

(Weatherwax Testimony, p. 15) However, because fire phenomenology is less well understood than are internal initiating events, and fire accident data are less than fully adequate, PRA has large uncertainties in this area. The point estimates of the contribution of fire to core melt frequency should be regarded as only very rough estimates that could easily be in error by a factor of 10 or more either way. (Buchbinder/Kubicki Testimony, p. 5) However, Staff witness Budnitz believes that only factors of 100 begin to be significant differences. (Budnitz, Tr. 7513)

1.1-107 IPPSS evaluation of fires was performed in a credible manner and appears to have identified the critical plant areas where a fire can cause an initiating event and simultaneously fail redundant safety systems which may be required to function in response to the initiating event. (Buchbinder/Kublicki Testimony, p. 4) Differences between Staff and IPPSS estimates of mean core melt frequencies for the sequences are small. (Id., p. 15)

(c) Wind

1.1-108 Extreme winds are not a significant contributor to risk at Indian Point Unit 3. (See IPPSS,

pp. 8.3-9 and 8.3-15, Table 8.3-3; see also Budnitz Testimony, p. 44; Peed Testimony, Table 2)

1.1-109 The most vulnerable critical structures at Indian Point have capacities of well over 100 mph, most over 200 mph. Forces associated with extreme wind sufficiently intense to cause damage to critical structures at Indian Point would be very rare. (Licensees 1, p. 49) Winds exceeding 100 mph at the site have mean occurrence intervals of 10,000 years. Winds exceeding 150 mph have mean occurrence intervals of over 100,000 years. (Budnitz Testimony, pp. 38-39) Such winds would cause widespread offsite damage as a direct result of the event before there would be any failures at Indian Point. (Licensees 1, p. 49)

1.1-110 While hurricanes were originally thought to be a significant contributor to risk at Indian Point Unit 2, a technical specification has now been implemented which requires the facility to shut down in advance of an approaching hurricane. (Rowsome III.A.4, p. 4) The criteria which trigger the anticipatory shutdown have been defined and included in the technical specification. (Id., p. 5) Offsite radiological risk can be largely eliminated by shutting down Indian Point 2 some hours ahead of

the arrival of a severe hurricane. (Id., p. 4) Accordingly, with the new technical specification in place, hurricanes no longer constitute a significant contribution to risk at Indian Point 2. (Id., p. 6)

1.1-111 Regarding tornados, the Staff reviewed the history of tornados in New York State. No tornado has ever crossed the Indian Point site. (Reed, Tr. 7563) Considering the probability that a tornado could cross the site, the IPPSS numbers constitute a fairly representative, if not conservative analysis of the tornado threat at Indian Point. (Reed, Tr. 7564) The analysis shows that tornados, while not negligible contributors to risk at Indian Point, are not a significant contributor. (Reed, Budnitz, Tr. 7572)

(d) Other External Initiators

1.1-112 The overall likelihood of floods from external sources capable of threatening the safety of the Indian Point plants would be once per 10,000 reactor years or less. (Budnitz Testimony, p. 49) Even if water levels were high enough to breach the plant's boundaries, core melt would not necessarily occur because the plant should be able to withstand this condition. Moreover, extra

protection would be provided because the plant would have been in a shutdown condition for some time if high water were threatening. (Budnitz Testimony, pp. 44-49) The frequency of water levels exceeding the grade elevation of the buildings at Indian Point is somewhere between once in a hundred million years and once in a trillion years. (Licensees 1, p. 45) Floods are not significant contributors to initiating core melt accidents, and their omission from full-fledged fragility analysis and quantitative risk analysis within IPPSS is justified. (Budnitz Testimony, p. 49; Licensees 1, p. 67)

1.1-113 The overall risk from aircraft crashes is much smaller than other risks to the public from the Indian Point plants. Therefore, omission of this category from full-scale risk analysis within IPPSS is fully acceptable. (Budnitz Testimony, p. 53; see Licensees 1, p. 67)

1.1-114 The risks associated with gas pipeline failures and barge accidents on the Hudson River are quite small compared to other risks posed by the Indian Point facilities, and therefore the omission of these events from a full scale risk analysis is acceptable. (Budnitz Testimony, pp. 59, 63)

C. PLANT RESPONSE

1.1-115 The IPPSS plant response analysis is particularly sophisticated and refined. (Weatherwax, Tr. 7919)

1. Data Base

1.1-116 For the data base used to model plant response, the authors of IPPSS collected industry-wide data and used it to provide initial probability distributions or generic distributions. These distributions were then supplemented with specific data from Indian Point. Four types of data were used: component failure data, maintenance and testing data, human error rates, and initiating event frequency data. Bayes' theorem was used to integrate the industry-wide data with the plant-specific specific data. (Licensees 1, pp. 59-60)

1.1-117 The authors of IPPSS obtained industry-wide data from NRC NUREG data summaries, IEEE STD-500, and the RSS. They obtained the data regarding Indian Point specific component failures from Indian Point LERs, control room equipment operating logs, component maintenance records, test records, and Indian Point station internal

event reporting documents. The Indian Point Units 2 and 3 component failure and unavailability distributions and the industry-wide data were used to quantify the system logic models. (Licensees 1, p. 60)

1.1-118 The authors of IPPSS did not model equipment performance under stress as though such conditions could simply be represented by a single, scalar factor. Instead the conditions under which each component or piece of equipment would need to operate were explicitly and individually assessed when equipment would be required to operate under conditions other than those for which they were designed and qualified. (Kaplan, Bley, Tr. 7033-7(.)

1.1-119 Maintenance and testing records for Indian Point were reviewed because system availability is reduced when components are removed for maintenance or testing. Thus identification of the frequency and duration of maintenance and testing activities is relevant. Industry-wide data for maintenance frequency and duration were based upon the equipment specifications, component normal service duty, the RSS, and experience with each component type. Unavailability data for Indian Point Units 2 and 3, combined with the industry-wide data, were

used to quantify the system logic models. (Licensees 1, pp. 60-61)

2. Modeling Techniques

1.1-120 The basic plant response model developed for internal initiating events in the IPPSS delineates the system functions necessary to avoid core damage and to protect the containment, and was used to calculate the probability that these functions are successfully provided. Because plant response to external initiating events is similar, this model was also used to analyze them. (Licensees 1, p. 52)

1.1-121 The IPPSS evaluated plant response to internal initiating events in terms of an overall event tree model supported by a detailed analysis to determine the reliability of the systems that must operate to accomplish the safety function identified in the plant event trees. The IPPSS considered system dependencies, including dependencies on auxiliary, shared, or connected systems. This analysis included consideration of states of electric power, and of the availability of auxiliary systems, interfacing systems, and actuation signals. The Indian Point FSARs, system descriptions, schematics,

piping and instrumentation diagrams, and operating and emergency procedures were used in the development of the system block diagrams and system fault tree diagrams to ensure accurate system modeling. The designers and operators of the plant also conducted an extensive review. The system fault trees were used to identify and display combinations of components whose failure could cause system failure. (Licensees 1, pp. 58-59)

1.1-122 FOE/Audubon is "sympathetic" to the use of fault and event trees if they are "carefully and thoroughly implemented." (Levi Testimony, p. 15) The Staff reviewed all of the event trees presented in IPPSS, and found the structure of most to be appropriate. However, certain changes were made in arriving at revised estimates. For example, Staff believed that the steam generator tube rupture event tree should include a sequence with a stuck open safety valve, and therefore modeled this sequence in their analysis. (Israel, et al., p. 9) Analyses by the Licensees have established that this accident sequence does not change the risk curves. (Richardson, Tr. 6447) In addition, the IPPSS ATWS event tree gave credit for an ATWS fix for which implementation has been deferred. (Israel, et al., p.9) Therefore, in Amendment 1 to the IPPSS, the contribution to risk from

ATWS without the modification was calculated and found to be negligible for the Indian Point plants. (Bley, Tr. 13021)

1.1-123 To analyze intersystem and intercomponent dependencies caused by human interactions, the authors of the IPPSS included human errors of omission and commission explicitly in the event trees and system cause tables, and selectively used the reliability tables of Swain's human reliability handbook to implement quantification. The starting point was the analysis of operation and maintenance procedures, which is especially important if operator action is required to effect actuation of a system or a collection of systems. They gained additional insights into the enumeration of actions by the simulation of event sequences using computer models. The authors of the IPPSS also considered possible incorrect judgments regarding the plant state and subsequent implementation of the wrong procedures. (Licensees 1, pp. 61-62)

1.1-124 The plant response review conducted by Sandia National Laboratories included an evaluation of IPPSS methodology and analysis assumptions, as well as a search for possible omissions. By reviewing the basic building blocks of IPPSS, Sandia was able to form its own

opinion regarding the correctness of IPPSS-designated dominant scenarios. (Israel, et al., p. 7) Therefore, the scope of this review goes beyond a mere examination of dominant scenarios. (Kolb, Tr. 7450-7451)

1.1-125 The Staff reviewed all the fault trees presented in IPPSS and found them generally to be a reasonable representation of the Indian Point safety systems. (Israel, et al., p. 8) However, both the Staff analysts and IPPSS analysts agreed that some changes were appropriate in the logic structure of the fault trees for the service water system, the auxiliary feedwater system, and the fan coolers. These changes were incorporated into the Staff's plant damage state frequencies (Israel et al., p. 8) and the results of the IPPSS Amendment 1. (Bley, Tr. 7085)

1.1-126 Regarding the main feedwater system, the authors of IPPSS were unduly conservative because they did not account for the ability of this system to remove decay heat following a reactor scram. (Israel, et al., p. 9)

3. Results of Plant Response Modeling - Core Melt Frequency

1.1-127 Core melt frequency is a poor indicator of public health risk. (Licensees 1, pp. 23-24, 67-58; Rowsome/Blond 5, pp. A-8-9, B-19; Richardson, Tr. 6394; Israel, et al., p. 16) There is no correlation between the magnitude of the core melt frequency and the magnitude of the risk. (Richardson, Tr. 6299) For example, 65 percent of the postulated core melt core scenarios at Indian Point 2 and almost 95 percent of those at Indian Point Unit 3 do not lead significant releases of radioactivity. Additionally, approximately 95 percent of the calculated early fatality risk is due to the interfacing systems LOCA, which contributes less than one-half of one percent to the core melt frequency. (Paddleford, et al., p. 8) On the other hand, core melt frequency is a useful indicator of economic risk to the customers and owners of the Indian Point plants, as it is a measure of the likelihood of losing the benefits of these plants. (Licensees 1, p. 68; Paddleford, et al., p. 8; Rowsome/Blond 5, p. B-19)

1.1-128 The Licensees estimated the median frequency of core melt at Indian Point 2 caused by

internally initiated events to be once in 20,000 reactor years and once in 7,100 reactor years for all initiating events (internal and external). The comparable values for Indian Point 3 are once in 33,000 and once in 20,000 reactor years.* (Paddleford, et al., p. 7) The Staff's best estimate of the core melt frequency is once in 3,000 reactor years for both plants when both internal and external initiating events are considered. The Staff acknowledged that their estimate, however, was conservative (more frequent) since they did not consider the strengthening of the control room ceiling panels nor did they eliminate the conservatisms in the frequency of hurricane-induced severe reactor accidents. (Rowsome/Blond 5, pp. B-18 to B-19) The Licensees did not take any credit for the adoption of administrative controls in the event of an approaching hurricane in estimating the contribution of such events to core melt frequency.

1.1-129 External initiating events such as wind and fire account for about one half of the contribution to the Indian Point 2 mean core melt frequency. A turbine

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* Although Licensees presented a slightly more frequent core melt value under Question One, Licensees' subsequent testimony under Question Five reflected these slightly less frequent values. The difference is inconsequential.

trip due to loss of offsite power is the most prevalent internal initiator. Small LOCAs and fire are the major contributors to the Indian Point 3 mean core melt frequency. (Licensees 1, pp. 69-70)

H. CONTAINMENT REPONSE ANALYSIS

1.1-130 Containment response analysis involves investigation of phenomena of reactor core melt accidents, and assessment of the response of the containment to these phenomena. (Licensees 1, p. 72; Meyer/Pratt Testimony, p. III.B-2) The authors of IPPSS evaluated degraded core phenomenology and containment capability, and developed detailed containment event trees. (IPPSS, p. 3.0-1; Licensees 1, p. 72) IPPSS containment response analysis, which includes a more comprehensive investigation of core melt phenomenology than that in the RSS, demonstrated that in general the Indian Point containment failure probability is much lower than had previously been assumed. (IPPSS pp. 2.1-2, 3.0-1) The major conclusions of the Licensees' containment analyses were substantiated by Staff's independent analyses. (Meyer/Pratt Testimony, p. III.B-62; see Findings 1.1-135, 1.1-145, 1.1-150 to 1.1-52)

1.1-131 Containment event trees, used in IPPSS and in Staff's analysis, are logic structures which provide an integrated and comprehensive analytical framework encompassing all important aspects of containment response. (Licensees 1, p. 80; Meyer/Pratt Testimony, p. III.B-16) The plant damage states derived from the plant analysis are the starting points for the containment event tree. (Licensees 1, p. 72; Meyer/Pratt Testimony, p. III.B-3) Branch points on the containment event tree represent success or failure probabilities based on physical laws, as well as on related experiments, analyses and engineering judgment. Detailed evaluations were performed by the authors of IPPSS and by Staff for those branch points which are most important to containment response. (IPPSS, pp. 0-52, 2.2-1; Licensees 1, p. 80; see Meyer/Pratt Testimony, p. III.B-16) The end point of the containment event tree is a set of radionuclide release categories which are used in the consequence analysis. (IPPSS, p. 0-38)

1.1-132 Over 6,000 accident progressions were considered in IPPSS. (Richardson, Tr. 7326; IPPSS Appendix 8.5.7, p. 8.5.7.1) Paths on the containment event tree were frequently carried through even when this required the modeling of phenomena that are physically

unrealistic. (Richardson, Tr. 7264; Liparulo, Tr. 7330-7334; Henry, Tr. 7336) These sensitivity studies provided justification for the assignment of probabilities to some unlikely paths on the containment event tree. (See Liparulo, Tr. 7730)

1.1-133 Both Licensees and Staff used the MARCH computer code supplemented by independent analyses and experimental information to quantify the containment event tree. In addition, Licensees used the COCO CLASS 9 and DEMO computer codes developed by Westinghouse. (Meyer/Pratt Testimony, p. III.B-3; IPPSS, pp. 2.5-1, 4.1-1 to 4.1-13)

1. Containment Strength

1.1-134 In assessing the strength of the Indian Point containments, it is important to distinguish between the onset of general yield of a primary structural component of the containment, and the ultimate failure pressure of the containment. The ultimate failure pressure is the pressure at which the containment would in fact fail and release radioactive material through a breach in the containment. (J. Meyer, Tr. 6791) The onset of general yield is the point at which the reinforcing steel in the

containment structure at its most limiting location begins to deform beyond its elastic limit. It has been shown to occur prior to ultimate failure and is therefore a limiting condition for which no loss of radioactive material occurs. (Bley/Richardson 2.1, p. 5; Toland, Tr. 7341-7342, 7345-7346, 7350-7351)

1.1-135 Both the Licensees and the NRC Staff performed independent calculations of that pressure at which there would be onset of yielding of the Indian Point containment structures. The Licensees calculated the point of onset of yielding for both Unit 2 and 3 containments to occur at approximately 141 psia, which corresponds to 126 psig. The point of onset of yielding was determined through a detailed analysis performed for the IPPSS using a methodology which accurately predicts the structural behavior of the containments up to their elastic limits. (Bley/Richardson 2.1, p. 5; IPPSS, p. 4.2-1; IPPSS, Appendix 4.4.1, p. 14-15; Licensees 1, p. 77-78; Richardson, Tr. 6396; Toland, Tr. 7340-41) The Staff also concluded that the onset of yielding occurs at 141 psia, based upon a study done for it by the Los Alamos National Laboratory. (NUREG/CR-2596; Meyer 2.1, p.3, fn; Meyer/Pratt Testimony, p. III.B-19; J. Meyer, Tr. 6821, 6823-6824)

1.1-136 UCS/NYPIRG performed no independent analysis of containment failure pressure. (Thompson, Tr. 6193, 6206) In addition, neither UCS/NYPIRG witness had reviewed the methodology that was used to determine the onset of yielding in the IPPSS or by the Staff consultants for the purpose of determining whether that methodology was appropriate. (Thompson, Sholly, Tr. 6208)

1.1-137 The onset of containment yielding would occur in the concrete reinforcing steel at the most limiting location. (Meyer/Pratt Testimony, p. III.B-19; Richardson, Tr. 6396; Toland, Tr. 7351; J. Meyer, Tr. 12519, 12531) IPPSS analysts found the limiting part of the containment structure to be in a location just below the connection of the hemispherical dome of the containment with the vertical wall of the lower cylindrical portion, because fewer seismic reinforcing bars exist in this area. The containment capability at 141 psia is not limited by the presence of penetrations or other discontinuities, but is determined by the expansion of the wall area on a broad scale due to internal pressure exerting a force on the wall. (IPPSS Appendix 4.4.1, pp. 14-15; Licensees 1, pp. 78-79) In the first 24 hours following accident initiation, the effects of pressure, temperature,

and radiation on piping penetrations, electrical penetrations, hatches, and major containment seals and ventilation valves would not cause them to fail prior to reaching the limit load criterion. (Licensees 1, p. 78; Toland, Tr. 7350-7351)

1.1-138 The controversy among the parties stems from their disagreement regarding the degree of uncertainty involved in determining that the containment could withstand a pressure of 141 psia. UCS/NYPIRG's position is that the failure pressure of the Indian Point containment is uncertain because the calculated numbers are based only upon analysis and no scale model tests to failure have been performed. (Sholly/Thompson Testimony, pp. 8-9)

1.1-139 In support of their assertion that there is considerable uncertainty surrounding the determination that the Indian Point containments can withstand the pressure of 141 psia, UCS/NYPIRG cited in their testimony a report prepared by Sandia Laboratories (Background Study and Preliminary Plans for a Program on the Safety Margins of Containments, NUREG/CR-2549, May 1982), which UCS/NYPIRG alleges raises issues affecting the uncertainty of calculation of containment failure pressure for the Indian Point facilities. (Sholly/Thompson Testimony, pp.

8-9) However, on cross-examination, witnesses Sholly and Thompson conceded that this NUREG dealt only with uncertainty in the calculation of failure pressures for nuclear containments and not with the degree of uncertainty associated with a calculation of the onset of yielding of those containments. (Tr. 6198-6199) During redirect examination conducted by counsel for the UCS, Dr. Thompson named another report done for the NRC Staff by the Massachusetts Institute of Technology (NUREG/CR-2898) which he testified also demonstrated an uncertainty with regard to the calculation of the pressure limits of the Indian Point containments. (Tr. 6254) However, Dr. Thompson conceded that NUREG/CR-2898 discussed only uncertainty in the ultimate failure pressure of containment, and not uncertainty in the onset of yielding. (Tr. 6264)

1.1-140 The Licensees and the NRC Staff agreed that there is great uncertainty in attempting to calculate the ultimate failure pressure for any containment structures. That is why both the Staff and Licensees calculated only the point at which there would be extensive yielding of the reinforcing materials in the Indian Point containments (onset of yielding), and assumed that is the point where the containment fails. (Bley/Richardson 2.1, p. 5; J. Meyer, Tr. 6791, 6824)

1.1-141 A pressure of 141 psia is a limit pressure, the upper bound on a pressure regime for which the structures behavior is well known, that is, how it will carry load, and how stresses will be distributed. (Toland, Tr. 7340-7342) In concluding that 141 psia was indeed the pressure at which onset of yielding would occur, the Licensees performed a deterministic analysis of the containment system, which included a review of all possible modes of failure, both structural and mechanical. (IPPSS Appendix 4.4.1, p. 2; Toland, Tr. 7340-7341) Secondary failure modes which might preclude the containment reaching 141 psia were sought, but none was found. (Toland, Tr. 7341) Manual calculations, with demonstrated accuracy from prior analyses, as well as special computer codes, were used in the containment capability evaluation. (IPPSS Appendix 4.4.1, p. 2)

1.1-142 This assumed failure pressure of 141 psia is 2.3 times the design basis accident pressure of 62 psia, and is approximately 30 to 40 psia greater than assumed for the Surry nuclear power plant containment in the RSS. (Licensees 1, p. 77; IPPSS Appendix 4.4.1, pp. 14-15; Bley/Richardson 2.1, pp. 5, 8) The factor of 2.3 times the design basis pressure corresponds to conserva-

tisms in the original containment structure design. Some of the factors contributing to the conservatism include application of load factors; no credit for strength of steel liner plate resisting accident loads; minimum strength of material considered in design basis accident evaluations rather than actual material properties; seismic rebar not considered to resist accident loads; and designer conservatism. (IPPSS Appendix 4.4.1, pp. 9, 15; Licensees 1, p. 78)

1.1-143 UCS/NYPIRG maintained that to have confidence in one's determination of the failure pressure of the containment one either had to perform scale model testing or add conservatism to one's analysis. (Thompson, Tr. 6193-6194) Although scale model testing has not been done to verify a failure pressure for the Indian Point containments, UCS/NYPIRG's concern seems to be satisfied by the conservative assumption of the Staff and Licensees for analytical purposes that the containment fails with the onset of yielding. (See Finding 1.1-140) Indeed, UCS/NYPIRG witness Thompson conceded on cross-examination that onset of yielding is more susceptible to analysis than ultimate failure pressure, i.e., that a calculation of the pressure at which the onset of yielding takes place is more reliable than the calculation of ultimate failure pressure. (Tr. 6197-6198)

1.1-144 While UCS/NYPIRG believes that additional conservatism should have been employed in assumptions as to the pressure at which the containment will fail, the testimony indicated that it is very important not to use overly conservative failure pressures. This tends to skew the results regarding the benefit of mitigation features. (J. Meyer, Tr. 6718)

1.1-145 The uncontroverted evidence demonstrated that the steel liner inside the containment was capable of extensive plastic deformation prior to failure, but would not fail at pressures below 141 psia. (J. Meyer, Tr. 6797-6799) The evidence therefore indicates that 141 psia is the lower limit for the actual ultimate failure pressure of the Indian Point containments. (Toland, Tr. 7340-7341; J. Meyer, Tr. 6822)

2. Seismic Capacity of Containment

1.1-146 Licensees presented testimony indicating that direct failure of the containment from a seismic event is impossible. (See Finding 1.1-83) Even accounting for uncertainty in the calculated median structural capacity, the containments would not fail at upper bound

ground accelerations of 0.8g. (Bley, Tr. 12990) The Indian Point containments could be subjected to no more than 0.8g acceleration from any earthquake that could affect the site, while the median structural capacity of the structure is 2.7g. (Perla, Tr. 13008) The largest ground acceleration expected within a 500-year period is less than 0.1g. (Licensees 1, p. 46)

1.1-147 Licensees' seismic expert testified that of ten containments analyzed by his organization only one has a higher seismic capacity than the Indian Point containments, namely Seabrook. (Wesley, Tr. 13001-13003)

3. Delayed Overpressurization

1.1-148 Gradual overpressurization failure is the most likely containment failure mode at Indian Point. (Meyer/Pratt Testimony, p. III.B-40) The median frequency of this failure mode accounts for approximately ninety-nine percent of containment failures. (Licensees 1, p. 89; IPPSS, Amendment 1, Table 8.1-1, p. 9.1-5) These failures are caused by the release of steam and by the release of noncondensible gases from the interaction of concrete with molton core materials. (Meyer/Pratt Testimony, p. III.B-13) The accident sequences leading to

gradual overpressurization failure are characterized by a loss of RCS inventory coupled with failure of both emergency core cooling systems and all containment heat removal systems. (Id., p. 10)

1.1-149 The IPPSS analysts calculated that late overpressurization failure of the Indian Point containments would not occur for at least twenty-four hours after the initiating event. (Liparulo, Tr. 7336-7337) The IPPSS was based on very conservative extrapolations which resulted in using a containment failure time of approximately 12-13 hours. (Licensees 1, p. 90; Meyer/Pratt Testimony, p. III.B-25; Liparulo, Tr. 7337, 7376; J. Meyer, Tr. 12557) Delays of 12 or more hours would provide time for reduction in the amount of fission products that could be released from the containment, as well as time for implementation of protective actions, if required. (Licensees 1, pp. 89-90) The pressure at which the containment fails would affect the timing of failure. However, because the authors of IPPSS used conservative extrapolations to calculate time to failure, even if one were to postulate containment failure at 120 psia for the accidents of interest, this would not occur until at least 12 hours after accident initiation. (Liparulo, Tr. 7338)

1.1-150 The combined mean frequency of occurrence of a gradual overpressurization failure for Indian Point Units 2 and 3 was determined in the IPPSS to be once in 14,000 reactor years. (Licensees 1, p. 111) The Staff's best estimate of the combined frequency of occurrence of the equivalent release category is once in 16,000 reactor years. (Meyer/Pratt Testimony, p. III.B-26) This low frequency results from a combination of the reactor coolant system design and containment design at Indian Point, which would prevent a breach of containment under minimal heat removal conditions. (Licensees 1, p. 24; Richardson, Tr. 7251) Only one of five fan coolers or one of six containment spray modes need be operable for these minimal heat removal conditions to be met. (Licensees 1, pp. 24, 72-73; Rowsome, Tr. 12929) With this minimum heat removal capacity available, maximum pressure would not even exceed the design basis pressure of 62 psia. (Richardson, Tr. 7251) Even if hydrogen burns and steam spikes are assumed to occur, if minimum containment safeguards are operable, pressures will not exceed 100 psia. (Id.)

1.1-151 The design of the fan coolers at Indian Point makes the potential for their failure from environmental conditions associated with a core melt accident

small. (Meyer/Pratt Testimony, pp. III.B-49 to III.B-50) The inoperability of the fan coolers and sprays has a negligible effect on the risk of early fatalities, but is more important for latent fatalities. (Id., p. III.B-49)

1.1-152 Neither the fan coolers nor sprays would be rendered inoperable by hydrogen burns. (J. Meyer, Tr. 12498, 12534-12535) Even if rendered inoperable, restoration of only one of the five fan coolers or one of the six containment spray modes, or other ad hoc recovery measures would insure containment integrity. (Bley/Richardson 2.1, pp. 6-7)

1.1-153 Although the Staff and Licensees agree that the Indian Point containments will not fail below 141 psia, they both developed failure probability curves for lower pressure levels. (Meyer/Pratt Testimony, pp. III.B-19 to B-20; Goeser, Tr. 7134; Richardson, Tr. 7340, 7342) The Staff's probability curve differs from that used in IPPSS for failure probabilities less than 80 percent because it approaches zero at approximately 120 psia while IPPSS curve approaches zero at approximately 136 psia. (Meyer/Pratt Testimony, pp. III.B-62, III. B-63)

1.1-154 Substantial capability exists in the containment liner to withstand deformation above 141 psia. (Richardson, Tr. 6396; J. Meyer, Tr. 6799) However, Staff extended the probability curve to a lower pressure (133 psia) because they believe that there would be extensive strain in the liner containment building at the intersection of the cylinder wall and foundation basemat which could cause local liner failure and leakage to the environment. (Meyer/Pratt Testimony , pp. III.B-19, III.B-62; J. Meyer, Tr. 6797-6799, 6825, 12518-12519) The Staff assumed a one percent per day leakage rate. (Meyer/Pratt Testimony, pp. III.B-19, III.B-24; J. Meyer, Tr. 6825) The impact on risk of this assumed leakage rate is negligible, even though this rate is an order of magnitude greater than the design basis leak rate. (J. Meyer, Tr. 12563-12564)

1.1-155 IPPSS analysis concluded that pressures below 141 psia do not produce excessive strain in the liner; thus there will be no loss of liner integrity below that level. In this analysis, IPPSS did not take credit for actual strength of the liner beyond its design strength. (IPPSS Appendix 4.4.1, pp. 9, 11, 14)

1.1-156 If the containment should fail due to overpressurization, the release would be in the form of a leak. The leakage that would occur at the weakest points of the containment above 141 psia would limit further pressure rises, thereby precluding a bursting type of release. (Toland, Tr. 7346)

1.1-157 UCS/NYPIRG asserted that the tight containment failure distribution used in IPPSS is critically important to the low probability of containment failure. (Weatherwax Testimony, p. 13) Staff stated that it is difficult to determine the impact of using the IPPSS containment failure probability curve instead of their broader curve. Use of IPPSS curve would result in postulated overpressurization failures occurring somewhat later and fewer hydrogen burns leading to containment failure than in their analysis. (Meyer/Pratt Testimony, p. III.B-62; J. Meyer, Tr. 12530-12531) However, Staff found that increasing the probability of overpressurization failure by 75 percent increased risk by only a factor of two. (Meyer/Pratt Testimony, p. III.B-48)

1.1-158 The authors of IPPSS testified that substituting the Staff's failure probability curve for the equivalent IPPSS curve would cause insignificant change in

the risk results. (Liparulo, Tr. 7335; Liparulo/Richardson, Tr. 7343; Richardson, Tr. 7324-7327) If all containment safeguards were operable, pressure within containment would not rise above design pressure. If none were operable, the pressure would eventually reach 141 psia, the point at which both Staff and the Licensees assume containment failure. (IPPSS, p. 4.0-4; J. Meyer, Tr. 6795-6796; Goeser, Tr. 7134; Richardson, Tr. 7251; Liparulo, Tr. 7335; Richardson, Tr. 7344-7345) Only a small fraction of the paths in the event tree lead to high pressure. (Richardson, Liparulo, Tr. 7326)

4. Steam Explosions

1.1-159 Most core melt accidents at Indian Point would not lead to early containment breach and, thus, early fatalities. (Licensees 1, p. 89) Steam explosions and missiles that could be generated by core melts were considered important potential contributors to early containment failure at the time the RSS was published. Since the RSS, additional research has demonstrated that reactor vessel geometry precludes the mixing of coolant and debris that is a prerequisite for large steam explosions, thus eliminating such explosions as important contributors to containment failure. (IPPSS, pp. 3.0-3 to 3.0-4, 3.1-31

to 3.1-57, 3.3-2 to 3.3-3; Licensees 1, pp. 24, 75, 89; Meyer/Pratt Testimony, p. III.B-15) Even if such debris/coolant interactions occurred, they would not produce loads sufficient to cause vessel or containment failure. (IPPSS §§ 3.2.1, 3.1.5; Licensees 1, pp. 75, 89; NUREG-0850, Appendix C)

1.1-160 Even though steam explosions are considered highly unlikely, they have been conservatively accounted for in the risk analyses presented. (Meyer/Pratt Testimony, pp. III.B-15 to III.B-16; Richardson, Tr. 7251) The Staff calculates the probability of a steam explosion failure to be very low, 100 times less likely than was reported in the RSS. (Meyer/Pratt Testimony, pp. III.B-15, III.B-20) UCS/NYPIRG agrees that the absence of steam explosions contributes to the low probability of containment failure. (Weatherwax Testimony, p. 13)

5. Hydrogen Generation

1.1-161 The amount of hydrogen required to be concentrated in the containment for hydrogen detonation and for a subsequent shock wave to breach the containment is not reached. (Richardson, Tr. 6398-6399; J. Meyer, Tr. 12498) Therefore, sequences involving hydrogen detonation were not modeled in IPPSS. (Richardson, Tr. 6398)

1.1-162 Staff and Licensees used conservative MARCH code calculations to estimate hydrogen generation for the time period between core uncovery and the movement of the degraded core to the lower reactor vessel, i.e., prior to core slump. Similar results for hydrogen generation during this phase were calculated by both parties. (Meyer/Pratt Testimony, p. III.B-52; IPPSS, p. 3.1-20) Their analyses differ regarding the additional amount of hydrogen that can be generated following core slump. (Meyer/Pratt Testimony, pp. III.B-52 to III.B-53) Licensees cited analyses of core melt phenomena which demonstrated that the core moves to the lower reactor vessel in a manner which limits debris/water reactions and resultant hydrogen generation. (IPPSS, p. 3.1-29)

1.1-163 The Licensees' analysis demonstrated that no more than 50 percent of the core material would be transported to the lower plenum in a short time interval. Despite this assessment, calculations were performed in IPPSS using up to 80 percent of the core material. (IPPSS, pp. 3.1-18 to 3.1-19, 3.3-2) Staff believes that there is uncertainty regarding the manner in which IPPSS modeled the movement of the core debris during core melt accidents. (Meyer/Pratt Testimony, p. III.B-53)

1.1-64 The amount of core debris available for oxidation, as well as the size of the core debris particles, affect the amount and rate of hydrogen generation. Licensees cited evaluations demonstrating that particles formed during the core melt process will be sufficiently coarse to rapidly cool the debris, thereby minimizing hydrogen generation. (IPPSS, pp. 3.1-9 to 3.1-14, 3.2-51, 3.3-1) Based on these analyses of core melt phenomena, the authors of IPPSS conservatively calculated that during core slump 50 percent of the still unreacted zirconium would react with water during high primary system pressure sequences and that 20 percent would react during low primary system pressure sequences. (IPPSS, pp. 3.1-21, 3.1-29, 3.3-1, 4.2-2; Meyer/Pratt Testimony, p. III.B-53) Staff stated that the particle size could be smaller than reported in IPPSS, leading to faster, more complete metal oxidation and thus more hydrogen generation. Additionally, Staff cited recent tests suggesting that up to 90 percent of the metal could be oxidized. Using this 90 percent estimation, coupled with their "lack of knowledge in this area," Staff assumed that 100 percent of the zirconium would be oxidized during in-vessel core heatup and meltdown. (Meyer/Pratt Testimony, p. III.B-54; Pratt, Tr. 12536-12537)

1.1-165 Staff and Licensees also presented differing analyses regarding the amount of metal oxidation that can occur as the core debris is released to the reactor cavity. Both parties used a similar bounding approach which maximizes the potential for overpressurization failure at the time of vessel failure. (Meyer/Pratt Testimony, p. III.B-55) The Staff analysis, however, resulted in predictions of greater amounts of hydrogen generation during this phase of the accident because Staff assumed that 100 percent of the core material would exit the vessel, thereby producing more hydrogen. (Id., pp. III.B-55 to III.B-56) The Licensees believe that this is not mechanistically possible. (IPPSS, p. 3.3-2) Although the Staff used the conservative bounding assumption that all the core material would be available for hydrogen generation, they concluded that the threat to the containment at this stage of an accident is minimal. (Meyer/Pratt Testimony, p. III.B-56)

1.1-166 The Staff's analysis differs from Licensees' concerning the coolability of the core debris in a flooded reactor cavity. Staff allowed one half hour more than the time they calculated for core material to cool when covered with water because of Biblis PRA (German risk study) date that indicates the possibility of a crust forming between debris and water, this delaying the cooling of

the debris. The assumption of an extra half hour cooling time results in an extra thousand pounds of hydrogen being generated. (Pratt, Tr. 12553-12554) IPPSS analyses showed that there would be rapid cooling of core material in a flooded reactor cavity, and that crust formation, if any were to occur, would be broken up by vigorous gaseous percolation. (IPPSS, pp. 3.2-50 to 3.2-54)

1.1-167 The Staff concedes that if IPPSS were to be right about rapid cooling of the debris bed, then there would be 1000 lbs. less hydrogen produced than in the Staff's analysis. (Pratt, Tr. 12553-12554) The Staff also believes that hydrogen combustion would be less likely to lead to containment failure if a coolable debris bed establishes within a half hour after vessel failure. (Meyer/Pratt Testimony, p. III.B-14)

1.1-168 The Licensees presented analyses indicating that for most of the analyzed accidents the hydrogen concentrations in the containment are not anticipated to be in the flammable range. (Licensees 1, p. 74) Other experimental and analytical data indicate that even when a hydrogen combustion is possible in the containment, such a combustion would not produce a breach of containment for a wide range of likely accident conditions. This is due to the amount of hydrogen available for burning, the manner

in which it burns and the capacity of the containment to sustain the calculated increases in pressure and temperature. (Licensees 1, pp. 74-75, 88-89) This is also due to the fact that hydrogen burns would not cause failure of containment cooling systems such as fan coolers or sprays, i.e., the probability that all fan coolers and sprays would be lost was found to be negligible. (J. Meyer, Tr. 12497-12498, 12505)

1.1-169 The Staff concurred that no subcompartments have been identified within the Indian Point containments that would cause non-uniform hydrogen concentrations leading to explosive mixtures of the gas. The circulation and mixing patterns have not been established well enough to preclude their existence. (Meyer/Pratt Testimony, p. III.B-57)

1.1-170 While electrical cables in containment could be ignited by a hydrogen burn, no equipment necessary to maintain containment integrity would be relying on those cables at that stage of an accident. (Pratt, Tr. 12500-12501) Recirculation spray pumps are located outside containment where associated cables would not be exposed to a hydrogen burn. (Pratt, 12534) Staff witness Pratt testified that this a very positive feature of the Indian Point design. (Tr. 12535)

1.1-171 The pressure surges from steam spikes or hydrogen explosion will almost always be less than 141 psia, making early containment failure very unlikely. (Bley/Richardson 2-1, p. 6) Hydrogen burns are not important contributors to releases at Indian Point. (See Meyer/Pratt Testimony, p. III.B-40) The Staff also testified that the contribution of hydrogen burns to the risk of early fatalities is small. (Meyer/Pratt Testimony, p. III.B-48)

1.1-172 The Staff performed sensitivity analyses to determine the importance of uncertainties regarding hydrogen-induced failure of the containment. They concluded that increasing by an order of magnitude the probability of a hydrogen burn causing failure of the Indian Point containments would have no effect on early fatalities, and would increase latent fatalities by only a factor of two. (Meyer/Pratt Testimony, pp. III.B-46 to III.B-51; J. Meyer, Tr. 12498, 12505)

6. Basemat Failure

1.1-173 The concrete basement beneath the reactor vessel would be subject to ablation by hot molten core debris coming into contact with it. For an event in

which there is an accumulation of debris in a dry reactor cavity, ablation of the basemat would occur. (Licensees 1, pp. 75-76) Staff believes that even with a dry cavity, complete basement penetration will not necessarily occur. (Meyer/Pratt Testimony, p. III.B-58)

1.1-174 For the dry-cavity case, the IPPSS authors assumed that the containment would fail by over-pressurization prior to basemat penetration. However, the Staff testified that the basalt-type concrete used in the Indian Point containment would limit the release of steam and noncondensible gases. Thus, they believe that an overpressurization failure caused by such releases would not necessarily precede a basemat penetration. (Meyer/Pratt Testimony, p. III.B-14) Because the health consequences of an overpressurization failure exceed those of a basemat failure, the Licensees have overstated the risk. (See IPPSS, at p. 6.5-1; Licensees 1, p. 91; Codell Testimony, pp. 4-5, 19-20)

1.1-175 In general, coolability of core debris would be influenced by how much and which way the debris is spread or dispersed. (IPPSS, pp. 3.0-5 to 3.0-7) The manner in which core material disperses at vessel failure depends on the pressure in the reactor vessel at the time of vessel breach, and on the reactor cavity geometry.

Substantial portions of the core debris are expected to be expelled or dispersed from the reactor cavity following a breach of the vessel for the more probable high reactor system pressure core melt events. The dispersed debris would likely be coolable due to its large surface area and the area of dispersal. The debris not dispersed would collect in the reactor cavity. (IPPSS, pp. 3.2-12 to 3.2-18; Licensees 1, pp. 76, 89) However, the basic plant geometry of the Indian Point units provides for the presence of water in the reactor cavity prior to a breach of the vessel, the continued availability of cooling water when containment safeguards are operational, and sufficient surface area for cooling. (Licensees 1, p. 76)

1.1-176 For events during which the reactor coolant system pressure is relatively low at the time of reactor vessel melt-through, almost all of the debris would remain in the lower reactor cavity area. The reactor cavity would have sufficient surface area to result in this debris being coolable. (Licensees 1, pp. 2, 76, 89) Therefore, in either the low or high pressure cases, i.e., whether the debris would be dispersed or not, the core debris would be coolable due to its large surface area, its particle size, and the availability of cooling water. (IPPSS, pp. 3.2-49 to 3.2-54; Licensees 1, pp. 76-77, 89)

1.1-177 If water is present on a continual basis, a coolable debris bed would form and no substantial concrete decomposition would be expected to occur. (Licensees 1, p. 76) The Staff believes that a coolable debris bed is not assured even with a flooded cavity. However, in that event the probability of basemat penetration would be no greater than 10 percent. (Meyer/Pratt Testimony, pp. III.B-22 to III.B-23, III.B-58) Furthermore, Staff states that overall risk is not affected by whether or not the basemat is penetrated. (Meyer/Pratt Testimony, p. III.B-49)

7. Release Categories

1.1-178 Both the IPPSS team and the Staff used the CORRAL computer code to determine the radiological release fractions to the environment based on the physical processes occurring in the containment building. (Licensees 1, p. 81; Meyer/Pratt Testimony, p. III.B-23) From these analyses, both parties developed a set of release categories. (Licensees 1, p. 82; Meyer/Pratt Testimony, p. II.B-24)

1.1-179 From this set, the Licensees identified two dominant release categories and their associated fis-

sion product release fractions. (Licensees 1, pp. 85-86) The release categories of interest are Category 2, the interfacing systems LOCA, which dominates early fatality risk and Category 2RW, gradual containment overpressurization, the most probable severe release category which dominates latent fatality risk. (Licensees 1, pp. 33-35)

1.1-180 According to Staff analysis, those release categories and their associated release fractions that dominate risk are A, B and C. (Meyer/Pratt Testimony, pp. II.B-25,III.B-37, III.B-40) Release category A is due to the large seismic event Beta Star, B is due to event V, interfacing systems LOCA, and all alpha failure modes, and C is due to all long-term overpressurizations and the steam generator tube rupture event. (Meyer/Pratt Testimony, p. III.B-37)

1.1-181 Staff states that the spectrum of release categories chosen bounds all possible releases. (Meyer/Pratt Testimony, p. III.B-24; J. Meyer, Tr. 12494-12495) Staff found they could have represented the releases with fewer than 9 categories. (J. Meyer, Tr. 12496)

1.1-182 Staff's release categories B and C are similar to Licensees categories 2 and 2RW, respectively.

(Meyer/Pratt Testimony, p. III.B-58) Staff's release category A is comparable to IPPSS release category Z-1Q. The Licensees testified that category Z-1Q is no longer possible at Indian Point. The Staff at the time of their testimony had not yet evaluated the Licensees' analysis. (Bley, Tr. 12990; Perla, Tr. 12992; J. Meyer, Tr. 12507-12516) Staff stated that if the analysis was correct, then release category A would be removed. (J. Meyer, Tr. 12509) There are no substantive differences between the Staff and the IPPSS analyses pertaining to these largest three releases. (Meyer/Pratt Testimony, pp. III.B-58 to III.B- 61)

8. Conservatisms In The Staff Containment Response Analysis

1.1-183 Staff witness Pratt conceded that there were excessive conservatisms in the Staff's hearing testimony and even stated that certain parameters should have been recalculated. Among the examples given were the chance of basemat penetration in the EF sequence, (Pratt, Tr. 12561-12562), and the split fraction for core melt quenching. (Pratt, Tr. 12548) Staff witness Rowsome conceded that the assumption that damage states TE and SE always lead to overpressurization failure was a conservatism that may have overstated risk. (Rowsome, Tr. 8844)

1.1-184 Staff employed the MARCH 1.1 computer code to analyze plant damage states instead of the MARCH 2.0 Code used by the Licensees (Pratt, Tr. 12518), and acknowledged that the MARCH 1.1 code had been the subject of criticism. (Pratt, Tr. 12538-12539) Staff witness Pratt conceded that he was one of the nuclear industry's most outspoken critics of the MARCH 1.1 code. (Tr. 12539) Staff was aware that its use of MARCH 1.1 resulted in more rapid core heatup than would actually be expected, and that MARCH 2.0 is an improved version. (Bernero, Tr. 12619)

1.1-185 Staff testified that it considers the NAUA-4 computer code to be the most valid code available for modeling aerosol behavior in the containment, and that the Staff uses it instead of the CORRAL code. Staff acknowledged that for late containment failure accident consequences with 8 to 10 hours holdup prior to release, a source term reduction of a factor of 10 results from use of the NAUA-4 code. (Bernero, Tr. 12617-12618) Yet, Staff employed the CORRAL code for its Indian Point calculations. (Meyer/Pratt Testimony, p. III.B-23)

1.1-186 Staff testified that the CORRAL code, the in-containment aerosol behavior model which was

employed in its Indian Point calculations, does not account for several important phenomenological factors, including: (a) plateout, (b) agglomeration, (c) solubility and capture of fission products in water, (d) the filtering effect of leakage from containment, and (e) these same factors for releases to the auxillary building. (Rowsome IV.C, p. 15)

1.1-187 It is therefore, unlikely that accident releases from the Indian Point units would be as great as suggested. We therefore find that the offsite radiological risks of severe accidents at the Indian Point units are likely to be less than Staff modeled them to be. (Rowsome IV.C, pp. 14, 16)

I. CONSEQUENCE ANALYSIS

1.1-188 The consequence analysis is an assessment of the potential offsite consequences and risks of various public health effects and economic impacts from postulated releases of radioactive material due to accidents at Indian Point 2 and 3. Those scenarios which lead to a release from containment to the atmosphere or which bypass containment were analyzed in great detail. (Licensees 1, pp. 17- 19, 26-26a, 91-109; Acharya Testimo-

ny; Codell Testimony) Each release scenario was assigned to one of several release categories based on the core and containment responses to the accident conditions. Accidents with the potential for release into ground water are lower in frequency and less severe in consequences than accidents which could result in atmospheric releases. Despite this fact, the Licensees modeled these releases as though they were atmospheric releases, thus overstating the risk. (Licensees 1, p. 91; Codell Testimony, pp. 18-20; IPPSS, pp. 2.2-10, 2.4-4, 6.51)

1.1-189 Meaningful consequence analysis of the Indian Point plants must be based on plant and site-specific data such as the characteristics of plant releases, meteorology, demography and topography, and protective measures such as evacuation and sheltering (Licensees 1, pp. 91, 105; Acharya Testimony, pp. III.C.A. 5-7)

1.1-190 The Licensees and Staff evaluated accident consequences and risks for the following indices: (a) early fatalities; (b) early injuries (radiation illnesses); (c) delayed cancer fatalities (excluding thyroid); (d) delayed thyroid cancer cases, fatal and non-fatal; (e) population dose (person-rem) to the exposed populations

(f) and off-site economic costs. (Licensees 1, pp. 26-26a, 98-104, 111-125; Acharya Testimony , p. III.C.A-3; Acharya/Blond Testimony, pp. III.C.B-7 to III.C.B-12)

1.1-191 In IPPSS, different probability distributions of selected health effects were combined with different frequencies to obtain a family of risk curves representing the range of uncertainty. (Licensees 1, pp. 93-94) The Staff utilized a somewhat different method of expressing risk by the use of point estimate risk curves which do not reflect uncertainty. (Acharya Testimony, pp. III.C.A-19, III.C.A-25 to III.C.A-29) The family of curves supplies the most information regarding risk. (Licensees 1, p. 15; Blond, Tr. 8627, 8682; see Findings 1.1-30. 1.1-346)

1. Analytical Procedures

1.1-192 PRA consequence analysis involves modeling the dispersion of radioactive material to the environment. The effects that those radioactive materials may have on the health of individuals or population groups depend upon the location of the material, whether it is airborne or deposited on the ground; its concentration; and the location and travel paths during evacuation of the

exposed population. The magnitude of the consequences of a postulated accident will vary with the timing and the duration of the radioactive release. The most important factors affecting this variation relate to weather conditions at the time of the release and during dispersion. Wind speed and direction, atmospheric mixing, and precipitation govern the area affected and the concentration of radioactive material at any given location. (Licensees 1, pp. 92-93; Pisello/Piccioni Testimony, pp. 3-6)

1.1-193 Protective actions which may be taken by the public can have an important influence on consequence estimates. This is particularly so with respect to early illnesses and fatalities because the health effects are not expected to occur unless specific threshold radiation doses are received. (Licensees 1, pp. 102, 110-115; Acharya Testimony, p. III.C.A-23; Sholly 1, pp. 4-5; Sholly, Tr. 8437-8438 ; Blond, Tr. 8677) In order to calculate radiation exposures and health effect probabilities, computer programs were used by both the Staff and the Licensees to construct the data bases and perform the analysis. (Licensees 1, p. 92; Acharya Testimony, p. III. C.A-2)

1.1-194 Both the Licensees and the Staff used modified versions of the Calculation of Reactor Accident Consequences (CRAC) computer program used in the RSS. The CRAC program was the first code developed for use in a comprehensive probabilistic assessment of the consequences of a reactor accident. It includes treatment of the effects of plume rise, wet and dry deposition, and changes in meteorological conditions (except wind direction) during plume transport. It also simulates the impact of evacuation and other mitigative measures, and it models doses and health effects. (Licensees 1, p. 94; Acharya Testimony, p. III.C.A-2)

1.1-193 Because the RSS was applied to a composite site, the authors of IPPSS developed an improved version of the CRAC program which allowed them to reflect fully conditions which are specific to Indian Point. This improved computer code is called Calculation of Reactor Accident Consequences Including Trajectories (CRACIT). (Licensees 1, pp. 94-95) The Staff also used the CRAC code modified in certain respects to apply to site-specific and problem-specific applications. The evacuation model was revised to allow for incorporation of site-specific delay time and effective evacuation speed as model parameters. (Acharya Testimony, p. III.C.A-2; Acharya/Blond Testimony, p. III.C.B-4)

1.1-196 The four principal differences between the CRACIT and CRAC codes are: (a) CRACIT models changes in plume trajectory, whereas CRAC models straight-line plume trajectories; (b) CRACIT models evacuation at a variety of speeds based on an analysis of the roadway network, whereas CRAC models evacuation radially away from the reactor site at a constant speed; (c) CRACIT is capable of modeling releases in as many as four separate phases (multiphase releases), whereas CRAC models all releases as a single "puff" and (d) CRACIT meteorological sampling consists of 288 random samples whereas CRAC combined 91 individual random samples with 16 direction sector samples, resulting in a total of 1456 scenarios for the calculation of probability distributions for consequences. Both Licensee and Staff applied these samples to individual release categories, and both used data from the same year. (Licensees 1, pp. 97-98; Sholly 1, p. 8; Acharya Testimony, pp.III. C.A-35, 36; Acharya, Tr. 8610; Blond, Tr. 8609-8610)

1.1-197 The CRACIT program has been used in a number of site-specific risk studies. (Potter, Tr. 7636-7637) A version of CRACIT, without the multiphase release component, was one of six codes exercised on a full range

of test problems in a recent international benchmark study of reactor accident consequence models sponsored by the Organization for Economic Cooperation and Development. The CRACIT code proved to be a state-of-the-art tool in this exercise. (Licensees 1, p. 95; Acharya Testimony p. III.C. A-35; Blond, Tr. 8604; Potter, Tr. 7635)

1.1-198 The CRACIT code used in IPPSS is more sophisticated than the CRAC computer code utilized by the Staff, and is a more "realistic" model particularly regarding long duration releases such as the Licensees' postulated 2RW release. (Acharya Testimony, pp. III.C.A-34 to III.C.A.-35; Blond, Tr. 8647, 8682)

2. Atmospheric Dispersion Modeling

1.1-199 In IPPSS consequence analysis, the distribution of airborne and deposited radioactive material in the environment was calculated as a function of time. Then the movement and shielding of people were taken into account, and the resultant doses and number of health effects calculated. As many as 300 weather condition scenarios were studied for a single postulated release to determine the probability distribution for each health effect analyzed. Assumed accident start times (month,

day, and hour) were randomly selected so that release start times were uniformly distributed over all months and over day and night conditions. Meteorological data were selected from the data base for corresponding times. This method of analysis is comparable to that used by Staff in its modeling of release sequences. (Licensees 1, p. 93; Acharya/Blond Testimony, pp. III.C.A-13, III.C.B-2, III.C.B-3

1.1-200 A screening process using the entire one-year meteorological data base was used to minimize sampling error and to ensure that low probability weather conditions, which would result in the most severe consequences, were specifically included. (Licensees 1, p. 107)

1.1-201 In addition to improved site-specific modeling and data, IPPSS authors employed a team of atmospheric dispersion experts, with years of experience with conditions in the Indian Point area, to evaluate the model results. (Licensees 1, p. 107; Potter, Tr. 7655-7657)

1.1-202 The following phenomena were modeled in the CRACIT dispersion calculations: (a) atmospheric dispersion of released materials; (b) wet and dry depositions; (c) plume rise; (d) inversion layers; (e) terrain model-

ing; and (f) long duration releases. (Licensees 1, pp. 95-98)

1.1-203 Atmospheric dispersion is the dilution and growth of plume dimensions. Wind speed determines initial dilution and travel speed, and wind direction and local terrain determine trajectory. As the plume is carried down-wind, its physical dimensions are influenced by atmospheric conditions. Changes in wind direction, wind speed, or stability during plume transport affect trajectory and dispersion. The capability to model variable wind direction trajectories was incorporated into the code. (Licensees 1, pp. 94-95; Potter, Tr. 7655-7657)

1.1-204 Wet and dry deposition are the mechanisms that remove isotopes from the plume, depositing them on the ground. Dry deposition occurs when some fraction of the particulates and reactive gaseous isotopes are absorbed on the ground. Wet deposition, occurring during precipitation, is more rapid and results in higher ground concentrations. (Licensees 1, p. 96; Palenik/Beyea Testimony, p. 77)

1.1-205 Plume rise refers to the height the released material reaches in the initial period following

release. Generally, the higher the rise, the lower the ground concentration near the point of release. The impact of plume rise diminishes with distance from the point of release. The wind speed, the energy in the release, and atmospheric stability determine the extent of plume rise. Each release category has a duration of release and release energy associated with it. Short duration releases with high energy content will rise higher than longer, low energy releases. (Licensees 1, pp. 96-97)

1.1-206 Inversion layers refer to the fact that during certain meteorological conditions, a layer of higher density air can act as a lid, thereby restricting the vertical rise of the plume or restricting vertical dispersion. The plume, however, may be sufficiently buoyant to penetrate the layer. (Licensees 1, p. 97)

1.1-207 Because of the manner in which containment would fail during an overpressurization accident, radiological releases occur over an extended period time up to several hours. (Toland, Tr. 7346) Staff agreed that it is more realistic to model these long duration releases as several separate time periods to account for changes in wind speed, direction and stability. (Licensees 1, pp. 97-98; Blond, Tr. 8647)

1.1-208 The important differences between the CRACIT and CRAC code analysis of atmospheric dispersion are the CRACIT capability for multiphase treatment of long-duration releases and the greater CRACIT meteorological sampling size. Other differences between the codes with respect to atmospheric dispersion analysis are not considered significant. (Acharya Testimony, p. III.C.A-35, III.C.A-36)

3. Dose Assessment Methodology

1.1-209 IPPSS employed the dose assessment methodology used in the Reactor Safety Study with only minor modifications. (Licensees 1, p. 98) Two basic elements of dose assessment are estimation of the concentrations of the radioactive isotopes in the environment, and identification and quantification of the exposure pathways for these isotopes. Time is an important variable in dose assessment because biological effects depend not only upon the total radiation dose, but also upon the dose rate. (Id.) Exposure may be internal, through ingestion of food or inhalation of air containing radioactive isotopes, or external, from isotopes that remain in the environment. Exposure can also be thought

of in terms of early and chronic phases. (Id.,98-99; Acharya Testimony, p. III.C.A-4)

1.1-210 Early phase exposure results from external gamma radiation from the radiation plume, internal doses from inhalation of radionuclides in the plume, and external gamma radiation from ground contamination due to atmospheric fallout. (Licensees 1, p. 99; Acharya Testimony, p. III.C.A-4) The IPPSS cutoff time for early phase exposure is the completion of the evacuation process, for those who evacuate, or 24 hours after the start of exposure for those assumed not to evacuate. (Licensees 1. p. 90) The Staff uses a tri-level approach to early phase exposure cutoff time. (Acharya Testimony, pp. III.C.A-7, III.C-2)

1.1-211 Chronic phase exposure results from ingestion of contaminated milk or crops and direct radiation from contaminated ground. Doses from inhalation of resuspended material are included in the computation, but are small relative to doses from other pathways. Chronic exposure would generally involve lower dose rates and longer time scales than would early exposure. (Licensees 1, pp. 99-100; Acharya Testimony, p. III.C.A-4)

1.1-212 Possible mitigative actions that can be taken to reduce early exposure are evacuation and sheltering. Sheltering reduces the dose incurred from inhalation, external plume exposure, and ground contamination. (Licensees 1, p. 99)

1.1-213 Measures such as interdiction and decontamination of land food stuffs such as milk and crops are two mitigative actions that can be taken to reduce long-term chronic phase exposure. The maximum decontamination factor considered practical is 20. (Licensees 1, pp. 100-101; Acharya Testimony, p. III.C.A-10 to III.C.A-11, III. C.A-37; Acharya, Tr. 8673) However, the area-weighted decontamination factor is about 5. (Potter, Tr. 7674)

1.1-214 A major distinction between the two exposure phases is that chronic exposure is entirely subject to administrative control. Measures such as interdiction of land would prevent doses caused by reoccupation of significantly contaminated portions of the plume exposure pathway. In theory all latent cancer cases from the chronic phase of exposure could be precluded by interdiction of all contaminated land and food. However, it was assumed that interdiction would not be applied to land contaminated at relatively low levels, that is, levels

which would lead to small increments in the risk of cancer fatality compared to background rates. The criteria for requiring land interdiction or decontamination was the same as that used in the Reactor Safety Study (RSS) which was adopted from the recommendations of the U.S. Federal Radiation Council and the Medical Research Council of Great Britain. (Licensees 1, pp. 100-101; Acharya Testimony, p. III.C.A-10)

4. Health Effects and Other Impacts

1.1-215 Both the Staff and the authors of IPPSS used the same health effects model as that used in the RSS. (Licensees 1, p. 101; Acharya/Blond Testimony, pp. III.C.B-7 to III.C.B-12)

1.1-216 These health effects can be divided into two categories, the first of which includes early and continuing somatic effects, such as fatalities and illnesses, that could occur after exposure to substantial doses of radiation in a short period of time. Although they would be expected to become manifest within days to weeks after exposure, illnesses and fatalities that could occur within a year of exposure are also included. These early and continuing somatic effects would not occur unless a signif-

icantly high threshold radiation dose was received. (Licensees 1, pp. 101-102)

1.1-217 The second category of health effects includes cancers, benign thyroid nodules, and genetic effects. Risk estimates for low doses are based on extrapolation of the known relationships between high radiation doses and probability of effect, because no adequate data exist for direct estimation of the incidence of risk of such effects from low radiation doses. Unlike the Intervenors, (Palenik/Beyea Testimony, p. 20), IPPSS risk estimates are approximately equivalent to mid-range estimates developed in 1980 by the Biological Effects of Ionizing Radiation (BEIR) committee in its most recent review. Virtually all of the cancer fatalities, thyroid cancers cases and man-rem as estimated in IPPSS for postulated severe accidents would result from low radiation doses for which risk estimates are based on extrapolation rather than direct experimental data. (Licensees 1, pp. 102-103)

1.1-218 Early fatality and early injury risk estimates generated by the CRACIT code are lower than those generated by the CRAC code because of more pessimistic delay assumptions used in the CRAC runs.

(Acharya Testimony, pp. III.C.A-9, III.C.A-36) The early fatality differences become pronounced only at the high consequence portion of the curves. Latent health effects generated by the two codes are approximately the same. (Acharya Testimony, p. III.C.A-41, Figures III.C.55 to III.C.59)

1.1-219 Supportive medical treatment would be given to people receiving potentially lethal bone marrow doses from early exposure (greater than about 175 rem) and minimizes early fatalities. The RSS estimated a national supportive medical treatment capability of 2,500 to 5,000 persons. (Acharya Testimony, p. III.C.A-3) With such treatment, the lethality threshold is elevated to about 300 rem to the total bone marrow. Although the Staff is in the process of updating the RSS estimate of the 2,500 to 5,000 figure for nationwide treatment, it has not yet completed its study. (Acharya, Tr. 8655) However, the recently published PRA Procedures Guide does not question the availability of supportive treatment below 10,000, suggesting that the RSS figures are underestimates. (Sholly, Tr. 8414) No special facilities are required for most persons needing supportive medical treatment. Only those persons receiving dose exposures exceeding 350 rem will need such special facilities. (Sholly, Tr. 8444-

8445) Medical treatment need not commence for 20 days to be effective. (Potter 5, p. 10; Sholly, Tr. 8412) This allows sufficient time to identify the affected areas and to screen the population that will require special treatment. (Potter 5, p. 10)

1.1-220 FOE/Audubon witnesses Drs. H. Jack Geiger and Victor W. Sidel testified regarding the difficulties in medical treatment of those potentially exposed to doses of radiation following a severe accident at Indian Point. (Geiger/Sidel Testimony)

1.1-221 Drs. Geiger and Sidel based their testimony on their understanding that a serious accident at Indian Point could result in large doses of radiation to residents of New York City metropolitan area specifically including parts of New York City itself. That understanding is based upon the Palenik/Beyea testimony (Geiger/Sidel Testimony, p. 2) The witnesses testified that, in their view, there is no logical or effective way of formulating an effective response to a serious accident at any nuclear plant site in an area of large population. (Geiger, Tr. 8482)

1.1-222 Geiger and Sidel expressed the view that in the event of a serious accident at Indian Point, it would be impossible to determine the population at risk since the meteorological data will simply not be rapidly available to determine which accident scenario is in fact being enacted. (Geiger/Sidel Testimony, p. 3; Geiger, Tr. 8482) The witnesses testified that computer modeling of radioactive plume would be unlikely to duplicate actual conditions and therefore make accurate plume tracking impossible. (Geiger/Sidel Testimony, p. 3) However, on cross-examination, the witnesses admitted to having no knowledge regarding the Indian Point station radiation monitoring instrumentation, plume tracking capability based on real time meteorological data or capabilities to predict changes in wind direction. (Sidel, Tr. 8543-8544; see Con Edison Onsite Testimony, pp. 21-24)

1.1-223 Although the witnesses testified that it would be impossible to perform triage should a serious accident occur at Indian Point (Geiger/Sidel Testimony, pp. 4-5), neither witness is a clinical specialist in the treatment of radiation injury. (Geiger, Tr. 8496; Sidel, Tr. 8500, 8542) They testified also that they are not radiation biologists nor are they experts in radiation dosimetry. (Sidel, Tr. 8542, 8548) Neither witness knew

whether there were any monitoring techniques to determine how much total radiation a person has received. (Sidel, Tr. 8543)

1.1-224 Staff analysis of the curves in section III.C.A of its testimony indicate that the risk of early fatality is about a factor of three (3) higher if supportive medical treatment is not administered. (Rowsome/Blond IV.A, p. 10; Acharya Testimony, Tables III.C.6 to III.C.7)

1.1-225 Additionally, testimony indicates that the early fatality risk is low, and that, with reductions in the source term, it would be reduced even further if not totally eliminated. (see Findings 1.1-407, 1.1-409, 1.1-302)

1.1-226 The Board finds that considering the time available to begin treatment and the fact that no special facilities are required for most persons needing this treatment, assuming no supportive medical treatment seems unduly conservative.

1.1-227 Although an analysis of the financial consequences of a reactor accident at Indian Point Units 2 and 3 was not included in the IPPSS, the Licensees included in their testimony an analysis and a family of

risk curves expressed in 1982 dollars to display this impact. (Licensees 1, pp. 26-26a, 103)

1.1-228 Both the Staff and the Licensees used the economic impact model from the Reactor Safety Study (see Section 12 of Appendix VI of the RSS). (Licensees 1, p. 103; Potter, Tr. 7638; Nesse Testimony, pp. 2, 4; Rowsome/Blond IV.A, p. 14)

1.1-229 The economic impacts considered in the development of the property damage risk curves include the cost of evacuation, relocation, interdiction, decontamination, and crop impoundment, but exclude costs associated with damage to the plant. (Licensees 1, p. 28; Rowsome/Blond IV.A, p. 14)

1.1-230 Both the Staff and the Licensees estimated genetic effects by applying a constant multiplier to the total man-rem exposure risk curve. This approach is consistent with the model used in the Reactor Safety Study. (Licensees 1, pp. 103-104; Potter, Tr. 7691-7692; Rowsome/Blond IV.A, p. 13; Sholly 1, p. 28) An explicit risk curve is not used to express genetic effects because there are different kinds. (Potter, Tr. 7692)

5. Plant/Site Data Used in Analysis

1.1-231 A meaningful assessment of the risk of the Indian Point plants cannot be made unless both plant-specific and site-specific data are considered. (Licensees 1, p. 91-92, 104-106; Acharya Testimony, pp. III.C.A-4 to III.C.A-7)

1.1-232 The consequence analyses presented by the Staff and the Licensees incorporated data that was specific to the plants and the Indian Point site and environs, such as meteorology, topography, demography, plant release characteristics, and protective measures such as evacuation and sheltering. (Licensees 1, pp. 91, 104-106; Acharya Testimony, pp. III.C.A-4 to III.C.A-7)

(a) Meteorology

1.1-233 Weather data from a representative one-year period, August 1978 through July 1979, were used in IPPSS analysis, and were derived primarily from the 400 foot tower located at the Indian Point site, which is instrumented at three levels. Wind speed, wind direction, atmospheric stability, precipitation, and inversion layer height were the parameters assessed. Inversion layer

height was determined from atmospheric soundings in the New York City area. Precipitation rate data were obtained for seven nearby locations, and wind data were obtained from 13 stations within 100 miles of Indian Point to determine plume direction as it travels away from the plant. Thus, any plume that is projected to leave the site area is assumed to be subsequently influenced according to weather data measured in an adjacent region. (Licensees 1, p. 104; Con Edison Onsite Testimony, pp. 19-25; Con Edison Supplemental Onsite Testimony, pp. 1-4; Acharya Testimony, pp. III.C.A-5 to III.C.A-6; L. Cohen, Tr. 11714-11738; Potter, Tr. 7655-7656)

1.1-234 UCS/NYPIRG criticized the plume transport modeling approach chosen by Licensees and Staff. UCS/ NYPIRG proposed that an alternative to a single year of meteorological data might be the creation of a "typical meteorological year" to represent the long-term average behavior of weather as discussed in the Sandia Siting Study. (Sholly 1, p. 27) The study of the dispersion meteorology in the vicinity of the Indian Point site has been extensive. It is the judgement of meteorologists that the data sources used in the IPPSS were sufficient to characterize appropriately the dispersion and the effects of local features on dispersion. (Potter, Tr. 7655-7657; L. Cohen, Tr. 11714-11738)

(b) Topography

1.1-235 To accurately represent the terrain data used in the three-dimensional dispersion model, a United States Geological Survey topographic map of the area was computer-coded and used in IPPSS model. Modeling the terrain enables simulation of terrain effects on plume trajectory and on dose at points along the trajectory. (Licensees 1, pp. 97, 105) CRAC appears to have modelled terrain differently. (Acharya Testimony, p. III.C-35) However, the Staff believes that the differences between CRAC and CRACIT in the treatment of terrain are not significant. (Acharya Testimony, p. III.C.A-36)

(c) Population

1.1-236 Estimates of 1980 population based on projections from 1970 census data were used for IPPSS analysis. (Licensees 1, p. 106) The Staff used an estimated 1990 population to reflect the approximate "mid-life" of the Indian Point site. (Soffer Testimony, pp. 2-3; Acharya Testimony, p. III.C.A-36) UCS/NYPIRG took issue with both the Licensees' and the Staff's use of population data, stating that neither approach completely reflects

the uncertainties involved. (Sholly 1, pp. 24-25) The Staff, however, reviewed both historical data and population projections beyond 1990. The historical data shows that Westchester County population declined between 1970 and 1980, and that Rockland, Orange and Putnam County populations all increased slightly. New York City population and that of the surrounding counties has declined significantly. Based upon the historcical data, the Bureau of Economic Analysis of the U.S. Department of Commerce has projected that for the area within about 50 miles of the Indian Point site, the population will either remain essentially the same or decline very slightly. Therefore, the 1990 year population is not only representative of the mid-life population, but a good representation of that which can be expected at the end of the service life of the Indian Point facilities. (Soffer, Tr. 8684-8685) Based upon the historical data and projections through the year 2010, the Licensees use of higher 1980 population estimates are slightly conservative because they result in overstatements of the societal consequences of severe accidents. (See Acharya Testimony, p. III.C.A-36)

(d) Plant Release Characteristics

1.1-237 In the IPPSS, separate CRACIT runs were made for each release category identified from the containment response analysis. The magnitudes of the radioactive releases were input as fractions of the initial core radioactivity that might leak out from the containment structure. In addition to release magnitude, the other parameters important to plume dispersion and consequence estimates were also considered, including: (1) time of release, (2) duration of release, (3) warning time for evacuation (warning time being the time estimated to elapse from the point at which the reactor operator determines that a release requiring emergency response would occur until the time the release does occur), (4) height of release and, (5) the energy content of the release. The Staff used the same parameters to develop their analysis. (Licensees 1, pp. 91-97, 105; Acharya Testimony, p. III.C. A-5)

1.1-238 The Staff used a core thermal power level of 3025 Mwt for Unit 3 and 2758 Mwt for Unit 2. The IPPSS used 3025 Mwt for both plants, which results in a slight overstatement of the Unit 2 consequence estimates. (Licensees 1, p. 105; Acharya Testimony, p. III.C.A-39)

(e) Emergency Response Assumptions

1.1-239 Portions of the consequence analysis set forth in Licensees' testimony assumed that protective actions would consist of evacuation of the 10 mile Emergency Planning Zone (EPZ), which would take place in accordance with the evacuation plan developed by Parsons, Brinckerhoff, Quade and Douglas, Inc. Other portions of Licensees' testimony assumed that sheltering would be the protective action taken. The evacuation consequence analysis provided for varying delays in the start of evacuation. For areas within 10 miles, five different scenarios were developed to account for time-of-day and seasonal variations in evacuation and population. The five scenarios are: (a) night; (b) week-school in session; (c) weekday-school not in session; (d) summer weekend or holiday; and (e) winter weekend or holiday. Each accident scenario was assigned appropriate population and associated evacuation data based upon the start time of the accident. Parsons, Brinckerhoff, developed evacuation trajectories and speeds, based on the roadway network in the EPZ. (Licensees 1, pp. 106, 121)

1.1-240 Delay time, the time between warning from the plant and the start of evacuation, is a key

parameter in computing early health effects. It was treated probabilistically by the authors of IPPSS. This treatment reflects the possibility that the entire evacuee population may be delayed longer than expected, as for example due to severe weather conditions, unavailability of transportation, or difficulties in communication, and that some segments of the population would respond less rapidly to the evacuation notification than would others. A range of delay times was modeled for every scenario. For some scenarios extra delay time was added, and some population groups in every scenario were delayed for additional periods. In 90 percent of the scenarios, the population was delayed from ½ to 2 hours; in 7 percent of the scenarios, delay times ranged from 1½ to 3 hours; and in 3 percent of the scenarios, delay times ranged from 2½ to 4 hours. The most likely delay time for most population groups in most scenarios was assumed to be one hour. However, it was assumed in all scenarios that at least 30 percent of the population was delayed more than one hour. (Licensees 1, p. 121; Potter, Tr. 12796)

1.1-241 Long delay times (5.5 to 13 hours) for populations in certain areas within three miles of the plant were assumed for the weekday school-in-session scenario because at the time the evacuation input data was

being established, reliance on public transportation was expected to require a second wave evacuation for some persons. Since that time, a new plan has been introduced by Westchester County Executive O'Rourke specifically to deal with concerns related to a second wave evacuations. (Licensees 1, pp. 116, 121; Potter, Tr. 12795-12797; IPPSS, pp. 6.2-8 to 6.2-10; see Finding (3.7-6 to 3.7-9) Based on the way delay time was modeled by the Licensees, the Board finds Mr. Sholly's criticism of the IPPSS with regard to the maximum delay time of 2 hours allotted for adverse weather conditions to be unfounded. (Sholly 1, p. 15)

1.1-242 Other consequence modeling performed by the Licensees assumed that sheltering would take place following an accident at Indian Point plant. The IPPSS took credit for a factor of two reduction in inhaled radionuclides as a result of sheltering. (Licensees 1, p. 117) UCS/NYPIRG took the position that the authors of IPPSS did not justify such a dose reduction factor. (Sholly 1, p. 22) However, Sandia Laboratories has concluded that a factor of two reduction in inhaled radionuclides can be used with "some degree of conservatism" and the NRC's PRA Procedures Guide appears to recognize a potential factor of two reduction in inhalation doses. (Id.)

1.1-243 The Staff chose three alternative sets of emergency response to bound the analysis, as well as to evaluate the sensitivity of the effectiveness of offsite emergency response to the chosen assumptions. (Acharya Testimony, pp. III.C.A-7 to III.C.A-10; Rowsome/Blond IV.B., pp. 3-5)

1.1-244 The first alternative emergency response assumption was designated by the Staff as "Evac Reloc". It assumed evacuation of the 10 mile radius EPZ, beginning two hours after warning with an effective radial evacuation speed of 1½ miles per hour. Beyond 10 miles, if the 7-day bone marrow dose remained less than 200 rem, it was assumed that relocation takes place 7 days after plume passage. Otherwise, it is was assumed to take place 12 hours after plume passage. (Rowsome/Blond IV.B, p. 4; Acharya Testimony, pp. III.C.A-7, III.C.A-9; Acharya, Tr. 8629, 8630)

1.1-245 The second Staff alternative emergency response assumption was referred to as "Early Reloc". It is identical to the Evac Reloc model except that everyone who requires relocation within the 10 mile EPZ is relocated 8 hours after plume passage rather than evacuated. (Rowsome/Blond IV.B, pp. 4-5; Acharya Testimony, pp. III. C.A-9 to III.C.A-10, III.C.2; Acharya, Tr. 8630-8631)

1.1-246 The third Staff alternative emergency response, was called "Late Reloc." It assumed that a severe earthquake or hurricane would preclude evacuation and would damage buildings so that no sheltering shielding factors were applied. The Late Reloc response in effect assumed that the exposed population were not already injured or killed by the damages buildings, and remained entirely outdoors until relocation, which was assumed to occur 24 hours after plume passage. (Rowsome/Blond IV.B, p. 5; Acharya Testimony, p. III.C.A-10; Acharya, Tr. 8631)

1.1-247 Since containment failure would not occur for at least 12 hours, and more likely 24 hours, after the initiating event, the Late Reloc assumption is comparable to assuming no emergency response at all for 36 to 48 hours after accident initiation. This response assumption is overly conservative and unreasonably overstates risk. (Potter 5, p. 6; Rowsome/Blond IV.B, p. 16; Acharya Testimony, p. III.C.A-33)

1.1-248 The IPPSS assigned an 11 hour warning time value to release category 2RW. (IPPSS, p. 6.2-54) Mr. Sholly testified that if containment failure were to occur 12 hours after an initiating event, an 11 hour

warning time would be overly optimistic because it is not clear that the operators would recognize that a sequence of events leading to core melt would be occurring within 1 hour of the initiating event. UCS/NYPIRG's position was that a 9 hour warning time would be more realistic, and recommended a warning time of 8 hours for modeling a 2RW late overpressurization scenario. Its witness testified that there would be a significant impact on the calculated consequences of a 2RW release if the warning time assumption was changed from 11 hours to 8 or 9 hours, since this would reduce the amount of time available for implementation of protective actions prior to breach of containment. (Sholly 1, pp. 19-21)

1.1-249 In the kinds of accidents that would proceed to extensive core damage, certain equipment or equipment functions would necessarily be unavailable early in the accident, and the current Emergency Action Levels for such conditions are very explicit in terms of the kinds of equipment failures that would obligate the Licensees to announce a General Emergency. The operators are not required to make a judgment as to how far the accident has progressed. If certain equipment functions were not available, the operators are required to promptly declare an emergency. (Bley, Tr. 7681-7682) Additionally,

the Licensees' testimony indicates that a 2RW release will likely not occur for at least 24 hours after the initiating event. In such a situation, if an 11 hour warning time were maintained for purposes of analysis, plant operators would then have at least 13 hours to recognize the possibility that a 2RW release would occur, and to recommend appropriate protective measures. (Licensees 1, p. 24; see Finding 1.1-149)

1.1-250 The Licensees' testimony presented sensitivity studies which demonstrate that early fatalities among evacuees would be prevented if evacuees began moving no later than the time of the release, and maintained a minimum speed of about 3 miles per hour. At a few miles from the plant, evacuations at this velocity could be initiated somewhat later without an increase in the risk of an early fatality. (Licensees 1, pp. 111-112) Finally, latent fatalities are not affected in a significant way by assumptions regarding evacuation time. (Potter 5, p. 13; Rowsome/Blond IV.B, p. 7)

1.1-251 The Board finds it likely that there will be more warning time than UCS/NYPIRG suggested. Even if UCS/NYPIRG's recommended warning time were adopted, plant procedures and reporting requirements are sufficient

to allow the operators to announce a General Emergency in a timely manner. Alternatively, the Board finds that there would be no appreciable impact on the calculated early and latent fatality consequences of a 2RW release even if an 8 hour warning time were assumed. Indeed, Staff analysis shows no early fatalities from Release Category C with 8-hour warning and Evac-Reloc response. (Acharya Testimony, pp. III.C-3, III.C-5)

1.1-252 UCS/NYPIRG testified that the Staff's Evac Reloc scenario for release category RC-C initiated by internal events is more or less comparable to IPPSS emergency response assumptions and that they would result in only a very small chance of early fatalities. (Sholly, Tr. 8428-8429; Acharya Testimony, Table III.C.5)

(f) Uncertainty in Consequence Calculations

1.1-253 For most of the parameters in the Licensees' consequence analysis, a single best estimate value was assigned. To reduce the uncertainties associated with these estimates, IPPSS authors improved the physical models for the transport and dispersion of release material. (Licensees 1, pp. 106-107)

1.1-254 Remaining uncertainties in these parameters and uncertainties in other aspects of the analysis are expressed as probabilities for each frequency versus health effect distribution for the significant release categories used in assembling the final risk curves. (Licensees 1, pp. 107-108) Separate CRACIT runs were made to produce each of the four sets of conditional risk frequency distributions. (Id., 108-109; see IPPSS Table 8.5.2-4a-e and Table 8.5.2-4-6)

1.1-255 Although UCS/NYPIRG claimed that the Licensees' testimony on uncertainties was only qualitative, it admitted that IPPSS included quantitative consideration of the net effect of uncertainties in many aspects of the dose calculations in considering modeling uncertainties. (Sholly 1, pp. 33-35) Staff characterized the approach used by Licensees as being in the forefront of modeling techniques. (Blond, Tr. 8614)

1.1-256 The Staff identified the following factors as the most important contributors to uncertainties in its analysis: (a) probability of the occurrence of an accident; (b) source terms; and (c) atmospheric dispersion modeling for the radioactive plume transport, including the physical and chemical behavior of radionu-

clides in particulate form in the atmosphere. (Acharya Testimony, pp. III.C.A-20 to III.C.A-21)

1.1-257 Those inputs which have a lesser effect on Staff's uncertainties include: (a) duration and energy of release, warning time, and in-plant radionuclide decay times; (b) meteorological sample scheme utilized; (c) emergency response effectiveness; (d) dose-conversion factors and dose-response relationships for early health consequences, including benefits of medical treatment; (e) dose-conversion factors and dose-response relationships for latent health consequences; (f) chronic exposure pathways, including environmental decontamination and the fate of deposited radionuclides; and (g) economic input data and modeling. (Acharya Testimony, pp. III.C.A-21 to III.C.A-25)

1.1-258 The Staff did not numerically bound the total uncertainties considered important although it did render some qualitative judgments with regard to all of the significant uncertainty factors. Staff stated that there are large uncertainties in its analysis. Furthermore, due to a probable overestimation of source terms, Staff states that it is more likely that they have overestimated rather than underestimated risk. (Acharya Testimony, p. III.C.A-25)

1.1-259 UCS/NYPIRG criticized the authors of IPPSS for performing sensitivity studies on CRACIT using a BWR-1 release category from RSS, and a 1930 MW BWR core inventory rather than a PWR core inventory. (Sholly 1, pp. 35-36) The BWR-1 release category source term is very similar to the PWR-2 release category source term and therefore is applicable for the initial test runs in which it was used. These sensitivity studies were performed to validate the operation of the CRACIT code by comparing CRACIT results to CRAC results using the same input assumptions. These tests involved insuring that the plume grid size was adequate, and that the CRACIT code was correctly performing the arithmetic involved in dose calculations. All other sensitivity studies used a release category which was determined specifically for Indian Point accident scenarios. (See IPPSS § 6.3)

J. EVALUATION OF PROTECTIVE MEASURES

1.1-260 The risk assessments presented by the Staff and the Licensees have identified the accidents which dominate the risk posed by the Indian Point plants. There are two: the gradual late overpressurization of containment (Staff release category RC-2, Licensee 2RW),

which is the most probable scenario, and the interfacing systems LOCA (Staff release category RC-B, Licensee release category 2), which would bypass containment, releasing radioactivity promptly to the environment. (Licensees 1, pp. 33-34; Rowsome/Blond IV.A, p. 8; Acharya Testimony, p. III.C.A-32)

1.1-261 Commission Question 1 asked for an assessment of the risk after any improvements identified in the response to Commission Question 4. No objective and useful way has been identified to translate the state of emergency planning and preparedness into emergency response effectiveness, that is, the speed and reliability of emergency response. Therefore, no quantitative conclusions can be drawn regarding the risk reduction potential of improvements in emergency preparedness. Although the risk reduction attributable to improvements in emergency preparedness is not known, it is expected to be small. The effects of various emergency response measures upon risk, however, have been calculated. (Rowsome/Blond IV.B, pp. 2, 3, 17)

1.1-262 Insights regarding measures to protect the public during emergencies such as the two scenarios above can be harvested not only from the risk assessments

performed by the Staff and Licensees but also from sensitivity studies of various alternate (to evacuation) measures. (Rowsome/Blond IV.B, p. 12; Licensees 1, p.110)

1. Slowly Evolving Accidents

1.1-263 With regard to the gradual late containment overpressurization scenario, the Licensees estimate that containment failure would not occur for a minimum of 12 hours and more likely 24 hours. (Licensees 1, p. 111; see Board Finding 1.1-149) The Staff has conservatively assumed 13 hours. (Meyer/Pratt Testimony, p. III.B-25) Given this time for evacuation, the Licensees do not expect any early fatalities among evacuees if the evacuees were to begin moving no later than the start of the release and maintain a minimum speed of about three mph. (Licensees 1, pp. 111-112) The Staff generally agrees that evacuation is highly effective for the more slowly evolving accidents except for those accidents initiated by severe earthquakes and hurricanes, where they pessimistically assume that neither evacuation nor early relocation is feasible. (Rowsome/Blond IV.B, pp. 7,11)

1.1-264 The need for evacuation of the public within the EPZ to reduce risk from slowly evolving acci-

dents has been estimated by the Licensees to be once in 14,000 reactor years; this is the combined mean frequency of IPPSS category 2RW scenarios for Indian Point 2 and 3. (Licensees 1, p. 111) The Staff's estimate of the combined frequency of slowly evolving accidents at Indian Point 2 and 3 is about once in 16,000 reactor years. (Meyer/Pratt Testimony, p. III.B-26) The frequency of this type of evacuation could be further reduced through plant recovery actions to prevent a release. (Licensees 1, p. 111)

2. Rapidly Evolving Accidents

1.1-265 For rapidly evolving accidents such as the interfacing systems LOCA, evacuation may not reliably clear the 10 mile EPZ ahead of plume arrival. (Rowsome/Blond IV.B, pp. 11, 16) These kinds of accidents represent the greatest emergency response challenge because they have the potential for releasing radioactivity relatively quickly to the environment. The Licensees selected this accident to study various alternative (to evacuation) emergency response measures. (Licensees 1, pp. 23, 116-125)

1.1-266 One such alternative measure would have the public take shelter, preferably in basements, until notified that the plume had passed. Those people in the plume pathway, as identified by radiological and meteorological measurements, would be notified where the plume boundaries are, and where and when to relocate. This alternative measure is referred to by the Licensees as "shelter first - relocation later." (Licensees 1, p. 122)

1.1-267 The travel distance to leave the affected area would depend on the width of the plume and the location within the plume of the population group of interest. As one moves away from the plume centerline, exposure decreases. Although the plume widens with atmospheric turbulence and also with distance from the point of release, the concentration of radionuclides, contamination levels, and resulting exposure will generally decrease. Therefore, the additional time needed to relocate outside a wider plume is largely offset by the lower contamination levels. In any event, the Licensees estimated that the maximum relocation distance from the centerline of the plume to its edge would be small -- 1.4 miles or less. (Licensees 1, pp. 122-124)

1.1-268 The Licensees performed a study in which they varied the amount of time during which people took shelter in their basements before being relocated following an interfacing systems LOCA. Two sheltering times, 6 and 24 hours, were compared to evacuation as modeled in the IPPSS. Even though the maximum relocation distance is small (from 0.09 to 1.40 miles), the Licensees conservatively assumed a one-half hour travel time (less than 3 mph evacuation speed). The analysis showed that the effectiveness of sheltering for six hours prior to relocation approaches that of the IPPSS evacuation model and exceeds that from sheltering for 24 hours. Sheltering for shorter periods prior to relocation and/or faster relocation times would improve the effectiveness of the shelter first - relocation later strategy, bringing it closer to the results achieved by the evacuation model assumed in the IPPSS. (Licensees 1, pp. 118-119, 122-125)

1.1-269 Staff performed a similar analysis for a category type 2 release (accidents not caused by a regional disaster) to compare their immediate evacuation approach (Evac-Reloc) to an approach which involves sheltering for eight hours after plume passage followed by relocation (Early-Reloc). Staff's results were essentially the same as those of the Licensees. That is, for this

type of accident, immediate evacuation may be only marginally more effective in reducing early fatalities compared to the early relocation approach. (Acharya Testimony, pp. III.C.A-31 to III.C.A-32; Rowsome/Blond IV.A, p. 8; Rowsome/Blond IV.B, pp. 6-7)

1.1-270 Because of the small area affected, the limited number of people requiring relocation, and the short travel distances, the shelter first-relocate later strategy would require less public transportation and present fewer logistical problems than anticipatory evacuation, which would entail moving large populations substantial distances. (Rowsome/Blond IV.B, p. 13; Licensees 1, pp. 25, 124-125, 131; Potter, Tr. 7642, 7643)

1.1-271 The need for evacuation of the public within the EPZ to reduce risk from a rapidly evolving accident such as the interfacing systems LOCA is extremely low, or about once in a 1,000,000 reactor years; this is the combined mean frequency of IPPSS release category 2, the interfacing systems LOCA for Indian Point 2 and 3. (Licensees 1, p. 111) The Staff's estimate of the combined frequency of such accidents at Indian Point 2 and 3 is about once in 1,100,000 reactor years. (Meyer/Pratt Testimony, p. III.B-26)

3. Further Insights

1.1-272 Licensees also presented an analysis using a very pessimistic emergency protection assumption in response to an interfacing systems LOCA. This analysis assumed that no protective measures would be taken for 24 hours after the start of exposure. People were assumed to spend some time outdoors, but most of their time was spent in a structure equivalent in shielding to a one-story frame or brick house. This limited shelter provides plume and ground shielding, but was assumed to provide no reduction in inhalation dose. This scenario is equivalent to an emergency response delay time of 24 hours during which even the simple precaution of sheltering in basements is not taken. Even under this "bounding" analysis, both Indian Point 2 and 3 meet the Commission's early and latent fatality safety goals. (Licensees 1, pp. 117-120)

1.1-273 Licensees also analyzed a scenario assuming that people from 0 to 50 miles of the plant would take shelter in a structure equivalent to the basement of a one-story frame or brick house for 24 hours and that the house was closed, thereby limiting infiltration of airborne radioactive material and inhalation dose. People

beyond 50 miles were assumed to take no protective measures. The Commission's early and latent cancer fatality safety goals are also met for Indian Point 2 and 3 for this scenario. (Licensees 1, pp. 117-120)

1.1-274 Using the IPPSS source term the Licensees investigated, the distribution of early fatality risk from all core melts as a function of distance from the site, assuming that there were no protective measures taken by the public for 24 hours. This investigation showed that essentially all the early fatality risk, more than 95 percent, is confined to a four mile radius of the site. Less than 0.2 percent of the early fatality risk is beyond 10 miles. Evacuation of areas beyond the 10 mile EPZ would at best be of marginal overall societal benefit. Based on this and reductions in the source term, a smaller EPZ may be justified. (Licensees 1, pp. 25, 112-113, 130; Potter, Tr. 7658 to 7659)

1.1-275 Justification for a smaller than 10 mile EPZ is further supported by analyses for the Indian Point plants and site, using assumptions identical to NUREG-0396, which show the likelihood of whole body doses of 200 rem or more resulting from core melt accidents falling off markedly well before ten miles. (Licensees 1, pp. 112-114, see Finding 1.1-388)

K. SOURCE TERMS

1.1-276 Staff and Licensee testimony focused on the amount and composition of radioactive materials that could be expected to be released to the environment in the event of a serious accident at Indian Point. This is commonly referred to as the "source term." (Stratton, et al., pp. 5, 15; Bernero Testimony, p. 2)

1.1-277 Determining an appropriate source term for use in analyzing the consequences of severe accidents is important, since overly conservative estimates of the magnitude of the source term released to the environment (i.e., overstatements of the quantities of radionuclides released) would lead to inaccurate and misleading overestimates of the consequences of various accident scenarios, and to inappropriate responses to these hypothetical accidents. (Stratton, et al., p. 5) In addition, the use of conservative source terms would overstate the value of the mitigative devices which have been considered in this proceeding. (Bernero, Tr. 12598-12599, 12601-12602)

1.1-278 WASH-1400 and its predecessor documents, e.g. TID-14844 and WASH-740, all made very conservative and bounding assumptions regarding the

composition and amounts of fission products which could escape from the core to the containment and subsequently to the environment in the event of a severe reactor accident. In particular, the assumption was made that, in case of an accident which involved breach of the containment, materials which could have been released from the core passed through the primary system into containment, and through the containment into the environment, without any significant reduction. (Stratton, et al., pp. 11-15) With respect to cesium and iodine, two of the most important radionuclides in determining short-term health effects from a severe accident, WASH-1400 assumed that these two substances remained in their elemental forms and passed without attenuation to the containment from the reactor vessel and primary system. (Id., p. 9) Since the IPPSS used basically the same assumptions as WASH-1400, the results in the IPPSS for some of the potentially very severe accident sequences show very little change from WASH-1400. (Id.)

1. Radionuclide Behavior

1.1-279 Both before and after the publication of WASH-1400, experimental programs suggested that there were many important physical and chemical processes neglected

by the authors of WASH-740, TID-14844 and WASH-1400 which would act to retain certain fission products within a damaged facility and thus prevent their release to the environment. (Stratton, et al., pp. 36-45; Bernero Testimony, pp. 3, 7)

1.1-280 Both the Licensees and Staff agree that a number of processes are potentially important in predicting fission product behavior following an accident and in preventing fission products from being released to the environment following their release from the core. These processes include: phase changes of the various fission product compounds and of the coolant; chemical reactions among the fission products and between fission products and the system; adsorption and desorption; solution in water; particle diffusion to surfaces through gas boundary layers; particle impaction on surfaces; diffusiophoresis; thermophoresis, which enhances particle wall plateout; sedimentation deposit on horizontal surfaces; particle-particle agglomeration; and, reentrainment from dry walls (Stratton, et al., pp. 31-33; Bernero Testimony, pp. 3,7)

1.1-281 The Licensees and Staff further agree, based upon analyses of releases during the Three Mile Island accident, and on confirmatory work done in the last

two years, that the previous assumptions in WASH-1400, i.e., that cesium and iodine would move as volatile materials directly from the core to the containment, are incorrect. Instead, the expected form of iodine will be cesium iodide, and not elemental iodine. (Stratton, et al., pp. 18-24, 28-29; Stratton, Tr. 8192; Bernero, Tr. 12584, 12604-12605) For example, the very low escape of iodine at TMI, compared to the escape of noble gases, was due to the relevant chemical state of iodine at TMI which was reducing, as opposed to oxidizing. That is, the iodine became iodide (I^-) and tended to form chemical bonds with hydrogen (H^+) or with metallic ions such as cesium Cs^+) or rubidium (Rb^+). Most of the metallic iodides are highly water soluble and, once in solution, do not become airborne and available for release. This chemcial explanation accounts for the behavior of iodine observed at TMI. (Stratton, et al., pp. 18-19; Stratton, Tr. 8192) The remainder of the active cesium which has not combined with iodine will form compounds with the hydroxyl ion or with oxygen. Cesium hydroxide is the most stable chemical form, second only to cesium iodide. (Stratton, et al., p. 29) Cesium iodide and cesium hydroxide are very soluble in water, and once in solution they stay in solution. (Id., p. 54)

1.1-282 The formation of cesium iodide, rather than iodine remaining in its elemental form, is important since many more of the processes listed in Finding 1.1-280 above would be operative in removing iodides than would be operative on elemental iodine. This will result in iodine being less mobile, i.e. less able to be released from the reactor coolant system in a core melt accident. It would also be less mobile once it reaches the containment building after leaving the reactor coolant system. (Stratton, et al., p.29; Bernero, Tr. 12584) This is important because cesium and iodine are among the most significant radionuclides for health risk purposes. (Bernero, Tr. 12591) Since iodine accounts for half of the potential early fatality risk, the formation of cesium iodide could have a significant effect on calculated offsite health effects. (Bernero, Tr. 12584)

1.1-283 Tellurium, a third element which is significant for health risk estimates, (Bernero, Tr. 12591), will probably be in metallic form following an accident because it has an extraordinary affinity for steel, iron and zirconium. It would deposit very quickly once vaporized from the core, and little would escape the core region. (Stratton, et al., pp. 54-56; Stratton, Tr. 8202)

1.1-284 The Licensees testified that additional confirmatory data supporting a source term reduced from that used in WASH-1400 and the IPPSS can be gained from an analysis of releases following accidents at the Windscale facility in England in 1957, the SL-1 reactor in Idaho in 1961, and TMI in 1979. (Stratton, et al., p. 46) Despite differences between these facilities and Indian Point, the accidents are illustrative of the physical and chemical processes which result in a small amount of radioactivity escaping to the environment relative to that in the core. (Id., pp. 46-56)

1.1-285 The Windscale facility was a graphite moderated, air-cooled, uranium core reactor. The air going through the core was sent up a stack and through a series of filters. There was no containment. The accident involved the burning of graphite and the core which, coupled with the lack of containment, allowed the escape of fission products from the core and stack. Although part of the reactor continued to burn for two days and almost all of the iodine in that part of the core was released from the fuel, only a fraction of the iodine exited the stack. (Stratton, et al., pp. 46-47; Stratton, Tr. 8217) Despite the limited similarity between the Windscale and Indian Point facilities there were similar-

ities in the individual physical chemical processes within the core. (Bernero, Tr. 12635)

1.1-286 The SL-1 reactor was a BWR test reactor with aluminum clad, uranium/aluminum alloy fuel. It was water moderated and was housed in a sheet metal building which had essentially no containment capability. (Stratton, et al., p. 47; Stratton, Tr. pp. 8218-8219) The accident there began when a control rod was suddenly withdrawn following a maintenance shut down. The increase in reactivity resulted in a power excursion and extensive core melting. About 19 percent of the fission product inventory was released from the core, but only 0.1 percent of the non-gaseous inventory reached the environment in the first two days, despite the fact that the sheet metal building housing the reactor was drafty and vented to the atmosphere. Calculations of the predicted releases from SL-1 using the WASH-1400 CORRAL code have been compared to the measured releases at the time of the accident. The actual releases were more than two orders of magnitude lower than the releases calculated by WASH-1400 methodology. (Stratton, et al., pp. 47-48) Staff witness Bernero testified that, although there are major differences between Indian Point and SL-1, one can learn something about the behavior of radionuclides such as iodine under

those accident circumstances that provide useful insight into the behavior of these radionuclides in accidents at large commercial reactors. (Bernero, Tr. 12636-12638)

2. More Realistic Source Terms

1.1-287 As a result of the post-TMI research into the magnitude of source terms for commercial reactors, the NRC ordered its Staff to review the matter itself. The results of the Staff review were published in NUREG-0772, "Technical Bases for Estimating Fission Product Behavior During LWR Accidents," dated June, 1981. In that document, the Staff acknowledged that the predominant chemical form of iodine was cesium iodide, and recognized the chemical and physical processes which would operate to reduce the amount of radionuclides exiting from a breach in containment. (Stratton, et al., p. 22; Bernero Testimony, p. 6; Bernero, Tr. 12591-12592)

1.1-288 NUREC-0772 concluded that releases from core melt events were likely to be substantially lower than presented in WASH-1400. (Bernero, Tr. 12593) However, the Staff continued to emphasize uncertainties and extreme conservatisms associated with highly improbable early failure accident sequences. (Stratton, et al., p. 22)

1.1-289 Peer review of NUREG-0772 was critical and a new, more in-depth analysis of fission product behavior is now being prepared by the Staff. (Stratton, et al., p. 22) This process involves the development of new computer codes to describe fission product transport in the core, primary system, and containment during a core melt event. Extensive experimental work continues to be performed to augment the existing data base. Using the new codes, a series of U.S. LWRs is being studied, one at a time, taking significantly different accident sequences and analyzing release and transport of radionuclides during those sequences. (Bernero Testimony, pp. 9-10)

1.1-290 The first of the plants, the Surry PWR, is being studied for a large break LOCA, small break LOCA, station blackout, and the interfacing systems LOCA (Event V). Dominant accident sequences for this type of plant will be identified and expected source terms developed for each dominant accident sequence. (Bernero Testimony, p. 10) Because Surry is different in terms of containment size and strength from PWRs such as Indian Point, the Zion plant, which is more nearly similar to Indian Point, will also be analyzed by the Staff. (Bernero, Tr. 12614)

1.1-291 A study similar to the NRC effort is being prepared by the industry degraded core group (IDCOR) in which Drs. Stratton and Rodger are participating. The IDCOR report is expected to be published this summer. (Stratton, et al., p. 23; Stratton, Tr. 8194; Bernero Testimony, p. 10)

1.1-292 An interim report is also expected this summer from the American Nuclear Society regarding its own review of the source term matter. Dr. Stratton chairs the committee preparing this report and Dr. Rodger is a member. (Stratton, et al., pp. 23-24; Stratton, Tr. 8194; Bernero Testimony, p. 10)

1.1-293 The International Atomic Energy Agency (IAEA) has completed a report which concluded that for accidents in which water or wet steam are in the flow path, or in which the containment remains intact or failure is delayed for several hours, the source term could well be reduced by several orders of magnitude as a result of the physical and chemical processes described above. The IAEA concluded that even a delay of "many tens of minutes" could afford an order of magnitude reduction. (Stratton, et al., pp. 24-25)

1.1-294 Based upon all of the previous research, experimental and accident data available, the Licensees presented revised estimates of more realistic (but still conservative) source terms as compared to those in WASH-1400 and the IPPSS. (Stratton, et al., pp. 6-9) These source terms are presented for the two release categories which dominate risk -- 2 and 2RW. In developing these source terms, attenuation for each isotope through various barriers was considered and a decontamination factor for each mechanism assigned. Available barriers between core and containment which were considered were:

1. Retention in the core material itself;
2. The primary system envelope:
   a. retention on primary surfaces,
   b. transfer into primary coolant;
3. Removal within the containment system
   a. by natural processes,
   b. by action of Emergency Core Cooling System; and
4. Retention on passing through the containment barrier into the environment.

(Id., p. 57)

1.1-295 The Licensees have detailed the development of their revised source terms for the release categories and for the six isotope classes considered. Taking as a starting point the IPPSS core releases shown in IPPSS Table 4.3, the following details are developed for each sequence:

1. Time elapsed since incident began;

2. Step-by-step description of an accident sequence (as described in the IPPSS) which leads to this release category;

3. Expected behavior of the particular isotope class during this portion of the sequence;

4. Fraction of the core inventory which is airborne in containment air; and

5. Fraction of the core inventory which escapes to the environment during each time interval.

(Stratton, et al., p. 58, Appendix A)

1.1-296 The Licensees presented revised source term estimates for release categories 2 and 2RW and compared them to the release fractions used in the IPPSS. For release category 2, that comparison shows, e.g., for iodine, cesium/rubidium, and tellurium/antimony, reduction factors of approximately 47, 50 and 300, respectively, from the release fractions used in the IPPSS. For release category 2RW, the respective reductions in release fractions for these same radionuclides are 6.6, 3 and 400. (IPPSS, Table 5.4-2, Stratton, et al., pp. 7-8, 59) These revised values are the upper limits on the amount of fission products which the Licensees' witnesses expect could escape from the plant. (Stratton, Rodger, Tr. 8194-8195; Rodger, Tr. 8223)

1.1-297 The physical and chemical processes upon which the more realistic source terms rely are always present. In fact, sedimentation alone supports the reduced source terms, even though the Licensees understated the effect of sedimentation by assuming much smaller aerosol particles than would be expected. (Rodger, Tr. 8195, 8234-8235)

1.1-298 The presence of water reduces the radionuclides available for release into the environment. Therefore, the more realistic source terms are understated because their calculation assumed absence of water and inoperable containment sprays. (Rodger, Tr. 8217, 8234; see Rodger, Tr. 8206)

1.1-299 Although Staff stated that TMI data are useful in evaluating what happens in the core and containment, it criticized reliance on TMI data as support for the more realistic source term, in part, because water was present during TMI. (Bernero, Tr. 12587-12588, 12643-12644) However, the Licensees' assumption of no available water results in a reduction factor of 100 and at TMI, with water present, there was an observed reduction factor of 1,000,000. (Stratton, et al., p. 7; Rodger, Tr. 8206)

3. Impact Of More Realistic Souce Terms On Risk

1.1-300 Four measures of accident risk were re-evaluated utilizing the revised source terms developed by the Licensees: early fatalities, latent facilities, man-rems and property damage. (Stratton, et al., p. 61)

1.1-301 Although other health effects indices are quite sensitive to reductions in source terms, early fatalities are the most sensitive to these reductions. (Bernero, Tr. 12611) To the extent that early fatality risk is overestimated due to overly conservative source terms, this overstatement is disproportionately large for densely populated sites. (Potter 5, p. 11)

1.1-302 The early fatality risk at Indian Point is due almost entirely to Category 2 releases. The Licensees' revised Category 2 source term is approximately equal to the IPPSS source term value divided by 50. Thus, even without any emergency response for 24 hours, no early fatalities would result from any serious accident at Indian Point. (Stratton, et al., pp. 6, 62; Potter, Tr. 8196) Assuming emergency response as modeled in the IPPSS, a factor of 15 reduction in the source term would effectively eliminate early fatalities. (Licensees 1, p. 131)

1.1-303 Even the use of the conservative IPPSS source term results in 95% of early fatalities being confined to within four miles of the Indian Point plants. (Stratton, et al., p. 62; see Finding 1.1-370) Based on early fatality risk, this fact, in combination with adoption of the Licensees' revised source terms, would provide justification for a smaller than ten mile EPZ, aswell as a review of the range of protective measures necessary to cope with reactor accidents. (Stratton, et al., p. 6)

1.1-304 Latent fatality risk at Indian Point is almost completely dominated by the 2RW slow overpressurization release category. (Stratton, et al., p. 63; see Finding 1.1-379) The Licensees' revised 2RW source term is approximately equal to a reduction in the IPPSS 2RW source term of about 20. The mean value for each plant for latent fatalities per year would be reduced by a factor of approximately seven using the Licensees' revised source term. (Stratton, et al., p. 63) This would render the health effects consequences of even a "worst case" accident similar to those of other large-scale industrial accidents. (Id., p. 6)

1.1-305 Utilizing the Licensees' revised source terms, the mean value of population dose expressed in man-rem is reduced by a factor of ten for both Indian Point 2 and 3. Since the worth of mitigative (and also preventive) devices are measured in terms of the man-rems averted, the Licensees revised source term also results in a reduction in the potential worth of any mitigative or preventive device by factor of ten for each plant. (Stratton, et al., pp. 63-64)

1.1-306 The property damage values for each facility would be reduced by a factor of approximately 17 by employing the Licensees' revised source terms. (Stratton, et al., p. 64)

1.1-307 Utilization of the Licensees' revised source term means that questions relating to the availability of supportive treatment and decontamination become much less important and in certain areas, e.g., in the New York City area, they become unimportant altogether. (Potter, Tr. 8196-8197)

1.1-308 The Staff agrees with the Licensees that the accident source term methodology employed in WASH-1400 results in overestimates of the quantity of fission

products released to the environment. This is because the WASH-1400 methodology neglects the physical and chemical processes which would afford retention of radionuclides in various parts of the damaged facility during a serious reactor accident. To the extent WASH-1400 methodology was employed, therefore, in the Staff and Licensee consequence analysis presented in the hearing, the Staff agrees that there are conservatisms in both analyses. (Bernero Testimony, pp. 3, 7; Bernero, Tr. 12582, 12626) The Staff has neither data nor information inconsistent with the Licensees' revised source terms. (Bernero, Tr. 12585-12592) The Staff cannot confirm that anything contained in the accident description and chronologies leading to the Licensees lower source term estimates is wrong. (Bernero, Tr. 12588-12589, 12620) While the Staff cannot confirm at this time that the Licensee's revised source terms are either right or wrong quantitatively, Staff witness Bernero testified that these revised source term reductions could be demonstrable. (Bernero, Tr. 12589, 12620)

1.1-309 Staff witness Bernero testified that the computer codes used by the Staff to evaluate the IPPSS source terms would have led to an overestimate of the source term. For example, in the analysis of the IPPSS,

the Staff used the MARCH 1.1 Code to model core heatup. This code predicts a more rapid core heatup than one would actually expect. (Bernero, Tr. 12619) The Staff is now using an updated version of the MARCH Code called MARCH 2.0, which gives an overall slower core melt time prediction (core heatup to core melt) than MARCH 1.1. (Id.)

1.1-310 In the analysis of IPPSS, the Staff used WASH-1400's CORRAL Code to predict radionuclide behavior in containment. The Staff is now using the latest version of the NAUA Code (NAUA-4) to do that analysis. (Bernero Testimony, p. 9) The NAUA Code appears to be the most valid model now available for prediction of aerosol behavior in containment. (Bernero, Tr. 12618) The NAUA Code predicts the fallout of aerosol at a more rapid rate than the CORRAL code. Therefore, delaying containment failure (which is the effect of using MARCH 2.0) results in a larger reduction in source term. (Bernero, Tr. 12617, 12591) This in turn results in a larger reduction in offsite consequences, and Staff witness Bernero testified that the effects can be "quite dramatic". For example the available evidence shows that delaying containment failure for eight hours results in a factor or ten or more reduction in the source term. (Bernero, Tr. 12591, 12617)

1.1-311 Mr. Bernero testified that all else being equal, the larger stronger containments would effect a greater reduction in source terms for slow overpressurization failures, since there is more time available for agglomeration, plateout, and gravity settling to occur. (Bernero, Tr. 12615)

1.1-312 Based on the available research and experimental data, Staff witness Bernero's personal prognosis is that there will be a reduction in overall releases on the order of a factor of ten. He testified that this would be a significant reduction since such a change in source term magnitude would have a significant effect on consequences of severe reactor accidents. (Bernero, Tr. 12589-12590, 12606, 12608)

L. OTHER CONSEQUENCE ASSESSMENTS

1.1-313 Other than the Staff and Licensees, two groups of parties presented estimates of consequences resulting from a severe accident at Indian Point. The New York State Attorney General, UCS/NYPIRG and the New York City Audubon Society presented the consequence estimates of Mr. Palenik and Dr. Beyea. (Palenik/Beyea Testimony

and Palenik/Beyea Supp. Testimony) FOE/Audubon presented consequences of a hypothetical release which was performed by Drs. Pisello and Piccioni. (Pisselo/Piccioni Testimony) Unlike the Licensees however, neither Palenik/Beyea nor Pisselo/Piccioni presented estimates of risk using PRA techniques, because neither made any attempt to address the probability of any of the scenarios for which they made consequence estimates. (Pisello, Tr. 8087; Palenik/Beyea Testimony, pp. 9, 25, 67-70) Only the Licensees and NRC Staff have quantitatively addressed the probabilities associated with the consequence estimates made by Palenik/ Beyea. (See Findings 1.3-1 to 1.3-17) No party has quantitatively addressed the probabilities of the scenarios presented by Pisello/Piccioni.

1. The Palenik/Beyea Consequence Estimates

1.1-314 UCS/NYPIRG witnesses Palenik/Beyea modeled potential numbers of latent cancer fatalities and thyroid nodules assuming the occurrence of one postulated accident scenario at Indian Point. They estimated that large numbers of such health effects would result from early phase exposure in the New York City area to a WASH-1400 PWR-2 release. (Palenik/Beyea Testimony, pp. 28-36)

1.1-315 Assuming such a release, Palenik/Beyea calculated latent cancer fatalities and thyroid nodules from a short-term exposure for a 50-mile sector encompassing a 7.5 degree wedge, assuming that the wind was blowing steadily towards New York City following the release. A range of latent cancer fatalities was calculated, ranging from 1,345, assuming one-day evacuation with no precipitation, to 64,130, assuming a four-day evacuation with precipitation. For thyroid nodules, the range calculated was 393,000 to 2,300,000, resulting in 3,500 to 22,000 fatal thyroid cancers. (Palenik/Beyea Testimony, pp. 30, 34)

1.1-316 The meteorological assumptions made by Palenik/Beyea in their consequence calculations included Pasquill D atmospheric stability class weather, a four meters per second (8 miles per hour) wind speed and a northerly wind. (Palenik/Beyea Testimony, pp. 30, 34; Beyea, Tr. 1945)

1.1-317 On cross-examination, Dr. Beyea testified that in order to achieve the results which he calculated for latent cancer fatalities and thyroid nodules in his testimony, conditions of Pasquill D

stability class, wind speed of 8 miles per hour and a northerly wind must occur simultaneously. (Tr. 3142) Dr. Beyea conceded that this combination occurs only approximately 2% of the time during the year. (Tr. 3143) Palenik/Beyea's use of a northerly wind was based upon an annual frequency of occurrence of a historical data base. (Beyea, Tr. 3138, 3141) This data represents the total number of hours in the year that the meteorological event occurred. However, annual frequency data represents only wind data for general climatological use, and the evaluation of plume transport would require the use of consecutive hourly data to define the true plume trajectory. (L. Cohen Supp. Testimony, p. 4) The 2% frequency of the stability class, wind speed and wind direction combination which were employed in the Palenik/Beyea final consequence calculations does not include the frequency of a constant wind speed (wind persistence) for four hours, which is approximately the amount of time necessary for an eight-mile per hour wind to reach New York City. Dr. Beyea did not know the frequency of the necessary combination of stability class, wind speed, wind direction, and wind persistence. (Beyea, Tr. 3155) In order to properly account for the parameter of wind persistence, Palenik/Beyea would have had to resort to records containing consecutive hourly data. (L. Cohen Supp. Testimony, p. 4)

1.1-318 Palenik/Beyea also assumed for purposes of their modeling of consequences that there was essentially flat terrain and that there were no obstacles between the Indian Point site and New York City. Therefore, they assumed that the site data could be applied downwind to the New York City area. (Beyea, Tr. 3149, 3170-3171) However, wind regimes between Indian Point and New York City can vary significantly. (L. Cohen, Tr. 11722) The proximity of certain topographic features south of Indian Point, namely High Tor Mountain, South Mountain and Hook Mountain State Park, will have an effect on winds flowing from the north. The topographic features in these areas range from 570-820 feet in elevation and thus form an obstacle to air moving south from Indian Point. Low level winds would have to diverge around the mountains, and, as a result, they are generally being reoriented with a more westerly wind component. Mountainous terrain not only changes plume trajectories but also increases the diffusive capacity of a plume from the induced mechanical turbulence generated by the air moving over rough terrain. (L. Cohen Supp. Testimony, p. 3) This importance of terrain in accounting for plume transport was addressed by the Licensees in the IPPSS. (L. Cohen, Tr. 11733)

1.1-319 Given the low frequency of occurrence of the meteorological conditions necessary for a plume to reach New York City and the topographic obstacles which would diffuse a plume heading from Indian Point southward, the probability of a plume reaching New York City is not substantial. (L. Cohen, Tr. 11733)

1.1-320 In converting his calculated doses to health effects, Palenik/Beyea surveyed the literature and selected a dose/health effect conversion factor range of from 50-500 cancer fatalities per million person rem. (Palenik/Beyea Testimony, Table 3, p. 30) •However, the 1980 BEIR report concluded that a dose/health effect conversion factor of 67-226 cancer fatalities per million person-rem was the appropriate range to use. (Id., Table 2, p. 20) Dr. Beyea conceded on cross-examination that the 1980 BEIR report is the most authoritative study on dose/health effect coefficients and that great weight must be given to the range selected by the BEIR committee. (Tr. 3072) Use of the dose/health effect conversion range recommended by the BIER committee would result in a factor of two decrease in Dr. Beyea's calculated maximum latent cancer fatalities. (Beyea, Tr. 3067-3068, 3175)

1.1-321 For thyroid nodules, Palenik/Beyea used a dose/health effects conversion range which was a factor of 4 to 5 higher than that used in the Reactor Safety Study. (Beyea, Tr. 3077-3078)

1.1-322 Palenik/Beyea stated that they were not testifying as medical experts nor as experts in the effect of radiation. (Palenik/Beyea Testimony, p. 17; Beyea, Tr. 2930, 3259)

1.1-323 In choosing a source term to use in his consequence calculations, Dr. Beyea used WASH-1400 assumptions with regard to the release fractions of radionuclides from a PWR-2 release. He assumed that 70% of the iodines of the alkalii metals would be released to the atmosphere in such an accident. (Palenik/Beyea Testimony, p. 25; Beyea, Tr. 3167) The witnesses testified that the source term is the overall key uncertainty in consequence analysis and if the source term were lower than they had assumed, the consequences would also be lower. (Beyea, Tr. 3211) Both the Staff and the Licensees testified that the source terms from WASH-1400 utilized by Dr. Beyea in his analysis were overly conservative, and the evidence indicates that at least a factor of 10 reduction in the WASH-1400 source terms is justified based on available data. (See Finding 1.1-308, 1.1-409)

1.1-324 Palenik/Beyea did not calculate the number of potential early fatalities from the postulated accident scenario which they considered because they testified that there were too many uncertainties in travel times and travel routes to do so. (Palenik/Beyea Testimony, p. 8) However, using the Parsons Brinckerhoff evacuation time estimates, (Id., p. 7, 79), Dr. Beyea determined that there would not be enough time in the event of a serious accident at Indian Point to prevent early fatalities within the 10-mile EPZ under certain circumstances. (Palenik/Beyea Testimony; Palenik/Beyea Supp. Testimony, Tables B1-B19)

1.1-325 Palenik/Beyea assumed that evacuation would begin at the same time as the release (Palenik/Beyea Testimony, p. 78; Beyea, Tr. 3123) and that therefore the evacuees would be leaving the EPZ through the plume. (Palenik, Tr. 3119)

1.1-326 Dr Beyea concluded that the longest warning time to avoid early fatalities is 4 hours under normal conditions and about 13 hours for the precipitation condition. (Palenik/Beyea Supp. Testimony, p. 59)

1.1-327 The Licensees and Staff's testimony indicates that, in the event of a gradual overpressure failure of the Indian Point containments (IPPSS Release Category 2RW, Staff Category C), the release of radioactivity would not take place for at least 12 and probably closer to 24 hours, leaving substantial time to begin evacuation before plume arrival and thus prevent any early fatalities. Because of the Indian Point containment strength, warning times in the range determined by Dr. Beyea to be necessary to avoid early fatalities will be available in the event of a gradual overpressurization failure of containment. (See Findings 1.1-149, 1.1-249)

1.1-328 For IPPSS Release Category 2, the other dominant Release Category identified for the Indian Point plants (Staff Release Category B), while early fatalities are possible, the Commission's early fatality safety goals are met for both Indian Point Units 2 and 3, even if there were no protective response action taken for 24 hours. Reductions in the source term anticipated by both the Licensees and the Staff would further reduce, if not eliminate, the early fatality risk. (See Finding 1.1-272, 1.1-302, 1.1-409) Finally, the probability of occurrence of a Release Category 2 event (an interfacing systems LOCA with containment bypass) is extremely low, (see Finding

1.1-369), as is the average risk of an early fatality for an individual living within one mile of the Indian Point site. That early fatality risk has been calculated by the Licensees to be about one in 140,000,000 per reactor year (7.1×10^{-9}) and about one in 160,000,000 per reactor year (6.4×10^{-9}) respectively for Indian Point 2 and 3.* (Licensees 1, pp. 31-32)

2. The Pisello/Piccioni Consequence Estimates

1.1-329 Drs. Pisello and Piccioni presented in their direct testimony four scenarios which they believe are probable consequences of a PWR2 type release from the Indian Point Unit 3 reactor. (Pisello/Piccioni Testimony, p. 3)

1.1-330 The first scenario postulated by Pisello and Piccioni addresses dose rates and doses in the immediate vicinity of the Indian Point plants under calm wind conditions. The release is postulated to be spread out in a full 360° circle from the release point. Under these conditions severe effects were predicted for workers

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* Licensees' Testimony (pages 31-32) incorrectly converted the value of 7.1×10^{-9} and 6.4×10^{-9}; the correct values are about 140,000,000 and 160,000,000 respectively.

at the plant site, and a dose rate of 5 rems per day is predicted at 100 meters from the release point even after 50 years. Pisello and Piccioni testified that under these conditions there would be an inability to perform maintenance on or surveillance of existing facilities at the plant site, which would increase the likelihood of the original accident moving into a second stage which could involve further releases from the damaged reactor, a major release from the second reactor, or one from the spent fuel storage areas. (Pisello/Piccioni Testimony, p. 4; Pisello, Tr. 8088-8089)

1.1-331 The second scenario postulated by Pisello/Piccioni assumes a constant three meter-per-second wind blowing toward New York City with no change in direction. They assume a constant deposition rate of 0.01 meters per second and calculate a dose rate of 9.6 rems per day after one day to 9mrems per day after 50 years. These dose rates were calculated for a point near the center of New York City. The witnesses calculated consequences of more than 1 million extra cancer fatalities as a result of such a release, assuming a dose to health effect ratio of 2,000 persons rem per fatality. (Pisello/Piccioni Testimony, p. 5; Pisello, Tr. 8089)

1.1-332 The third scenario postulated by Pisello and Piccioni involves a plume which moves Northwest from the site over the Catskill catchment area, which provides most of the water supply for New York City. The witnesses then postulate a rain storm which washes the contaminants out of the plume, contaminating the New York City water supply, and predict that New York City drinking water would sustain concentrations of radionuclides in excess of the Maximum Permissible Concentration (MPC) for two and one half years. (Pisello/Piccioni Testimony, p. 6; Pisello, Tr. 8089)

1.1-333 The final scenario postulated by Pisello and Piccioni deals with a plume moving in a direction which keeps it over land for a considerable distance. The witnesses calculated that 10,000 square miles of New York State would require interdiction of agricultural land in this scenario and that 72% of the contaminated agricultural land would lie outside the 50 mile ingestion pathway EPZ. Ninety thousand latent fatalities are predicted by the witnesses to result from this scenario. (Pisello/ Piccioni Testimony, p. 7; Pisello, Tr. 8090)

1.1-334 In all their calculations, Pisello and Piccioni used a simplified wedge model which they charac-

terized as most suitable for calculating surface deposition of radioactivity outside a radius of 10km from the release point. (Pisello, Tr. 8153) Given the fact that scenario one in their testimony concerns a release which remains in the immediate vicinity of the plant site, the witnesses conceded that they could not defend the use of the wedge model for scenario one calculations. (Pisello, Tr. 8109)

1.1-335 The witnesses' testimony did not include any criticisms of the manner in which the consequence analysis was conducted either in the IPPSS or by the NRC Staff's consequence analysts. On the contrary, the witnesses testified that the CRACIT consequence modeling used by the Licensees in the IPPSS was more sophisticated than the simplified wedge model that they used (Pisello, Tr. 8161) and that in using the simplified wedge models for each of the four postulated scenarios, they were just trying to get a simple, quick representation of what one might expect in a reactor accident. (Pisello, Tr. 8109) Indeed, Dr. Pisello conceded that the simplified wedge model which he used does not accurately represent the distribution of airborne radioactivity in the armosphere. (Tr. 8108) Prior to testifying, the witnesses had spent only a few hours reviewing the IPPSS. (Pisello, Piccioni, Tr. 8099-8100)

1.1-336 In using the simplified wedge model, predictions of consequences can change substantially by varying some of the parameters. (Pisello, Tr. 8154) For example, the consequence predictions vary inversely with the mixing height and wind speed assumed. (Id.)

1.1-337 For scenario one, the witnesses made no assumptions with respect to either the size of the breach in containment, the temperature of the release or the pressure within the containment at the time of the release. (Pisello, Tr. 8102-8103) The meteorological data used by the witnesses was Dr. Pisello's personal experience on Broadway in Buchanan, New York on summer evenings. (Pisello, Tr. 8106) No frequency analysis was made by the witnesses of the combination of calm conditions assumed for scenario one and the 100 meter mixing height used. (Pisello, Tr. 8107) The meteorological data consulted by Pisello and Piccioni indicates that a lid height between zero and one hundred meters (the mixing height assumed by the witnesses for scenario one) occurs only 1.4% of the time for the geographical area from which their data was taken, which is Fort Totten on Staten Island. (Pisello, Tr. 8166, 8188) The witnesses made no attempt to model plume buoyancy. (Pisello, Tr. 8109)

1.1-338 With regard to scenario two, which postulates serious consequences for the New York City area, Pisello and Piccioni made meteorological assumptions which did not account for either wind shear effects or changes in wind direction. (Pisello, Tr. 8116, 8118) In addi- tion, in modeling the travel of the plume from Indian Point to New York City, the witnesses did not account for topography beyond the observation that there are no mountain ranges beween the Indian Point site and New York City. (Pisello, Tr. 8118, 8119) This assumption is not correct. (See Finding 1.1-318) Further, the witnesses assumed no shielding reduction for anyone in the path of the plume. (Pisello, Tr. 8122-8124) On cross-examination, the witnesses could not refer to any other study of long-range ground dose which assumed no shielding reduction. (Pisello, Tr. 8125) Dr. Pisello recognized that a shielding factor must be applied. (Id.) Finally, the witnesses made no allowance for decontamination efforts in the New York City area following their postulated release. (Pisello, Tr. 8128-8129)

1.1-339 With regard to scenario three, in contrast to Pisello/Piccioni's admittedly quick and simple analysis, (Pisello, Tr. 8109), the Staff performed a

detailed analysis of the potential for accidental contamination of the New York City water supply system from an airborne release of radionuclides at Indian Point. (Codell Testimony, pp. 24-29)

1.1-340 Staff witness Codell first developed an empirical model relating the concentrations of Sr-90 and Cs-137 in New York City water to the quantity of radionuclides deposited on the land surface. He used published data on radioactive fallout deposition and tap water concentrations in New York City to adjust the parameters of his model. (Codell Testimony, p. 26) The analysis was ultimately restricted to Sr-90 because it would be responsible for 80-90% of the long term whole body dose, and neglecting all of the radionuclides would lead to only a small error in drinking water dose estimates. (Id., p. 27)

1.1-341 Dr. Codell performed a modified CRAC code analysis which assumed a Category C (or in Licensees' terminology, a 2RW) release. A Category C release was used because that event alone would account for about 98% of the airborne/liquid pathway risk. (Codell Testimony, pp. 27-28) Dr. Codell used the CRAC code data from a Category C event in a computer program which factored

water shed dimensions, distance and the wind direction probability (wind rose) to predict the cumulative frequency distribution of Sr-90 deposition on the water sheds for the Croton, Catskill and Delaware systems which serve New York City, either separately or combined. The Sr-90 deposition was then used to calculate the cumulative frequency distribution for New York City tap water concentrations using the Empirical Concentration Model from which dose estimates could be made. (Id., p. 28)

1.1-342 For the combined Delaware/Catskill subsystem, which provides most of the drinking water for the New York City area (91%), the Staff's calculation shows that there is only about a 1.1% chance of exceeding the MPC for the first year for Sr-90. (Codell Testimony, p. 31; Codell, Tr. 8660)

1.1-343 Finally, Dr. Codell testified that because of the proximity of the New York City reservoirs to the Indian Point site, there is not likely to be any surface water supply which could become highly contaminated from an accident at Indian Point. (Codell Testimony, p. 33)

1.1-344 With regard to Pisello and Piccioni's scenario four, which postulates a large land area interdicted as a result of a serious accident at Indian Point, although the witnesses simple wedge model predicted an area requiring interdiction far larger than the 50 mile ingestion pathway EPZ, the Staff's detailed economic analysis did not include interdiction even as far out from the site as the New York metropolitan area (approximately 35 miles) because the Staff determined the likelihood of having to implement interdictive measures at that distance to be so low as not to warrant inclusion. (Nesse, Tr. 8668)

1.1-345 In performing their health effects calculations, Pisello and Piccioni used a dose-to-health effects conversion factor of 2,000 person rem per fatal cancer. This is a lower fatality threshold than calculated by either of the two models discussed in the BEIR-1980 Committee Report. The BEIR Committee calculated a dose-to-health effects conversion factor using both a linear model and a linear quadradic model. (Pisello/Piccioni Testimony, Table VII) While the Pisello/Piccioni cancer fatality threshold is even lower than that calculated using the linear model, the BEIR-1980 Committee actually favored the linear quadratic model, which results in a

much higher cancer fatality threshold and thus fewer health effects than that utilized by Pisello/Piccioni in their calculations. (Pisello, Tr. 8140-8141) The linear quadratic model favored by the BEIR- 1980 Committee calculates a dose to health effect figure of 6,300 person rems per fatal cancer. Using the linear model also reported in the BEIR 1980 study, a dose-to-health effect figure is 2,600 person rems per fatal cancer. (Pisello/Piccioni Testimony, Table VII)

M. SUMMARY OF RISK RESULTS

1.1-346 Both the Licensees and Staff presented probabilistic risk assessment results.* The Licensees' results were presented principally as a family of risk curves which are a complete display of risk, including uncertainty. (Licensees 1, pp. 15, 19-20, 26-26a; IPPSS Amendment 1, pp. 8.1-6, 8.1-7) Such risk curves provide the most information regarding risk. (Blond, Tr. 8627, 8682; Rowsome, Tr. 8813) The Licensees' risk curves for

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*Staff presented both "before fix" and "after fix" risk values. Because the fixes have already been implemented, only the "after fix" values are of significance here. All of the Staff results presented in this section are drawn from Staff curves labelled Evac-Reloc + Late Reloc.

Indian Point 2 and 3 are attached as Appendices A and B to these Findings.

1.1-347 Because the Staff does not quantify uncertainty (Acharya Testimony, p. III.C.A-25; Blond, Tr. 8617, 8682), their risk curves are "best estimate" (point estimate) plots. (Acharya Testimony, p. III.C.A-39 n*) Staff stated that, based on a qualitative assessment, the uncertainty bands accompanying their results are large, and are much greater below than above their point estimates. That is, it is more likely that they have overestimated rather than underestimated risk. (Rowsome IV.C, pp. 14, 18-19; Acharya Testimony, p. III.C.A-25)

1.1-348 Point estimates are not precisely defined; they are not means, medians, or modes. (Kaplan, Tr. 8247) Therefore, the relationship between Staff's and Licensees' expression of risk results is not clear, and direct comparisons may not be appropriate. (Acharya Testimony, pp. III.C.A-39 n*, III.C.A-43 to III.C.A-45)

1. Early Fatality Risk

(a) Societal Risk

1.1-349 For the most severe accidents, there is a very small probability that sufficient radioactivity could be released to cause potentially lethal exposures. (Rowsome/Blond IV A, p. 9) The risk of early fatalities from reactor accidents at Indian Points 2 and 3 has been found to be extremely small, and only results from accident scenarios with a prompt release of radioactive material. (Licensees 1, p. 23) The Licensees have constructed early fatality societal risk curves for reactor accidents at Indian Point 2 and 3. (Licensees 1, pp. 26-26a (Figures III-1a and III-2a)) (Appendices A and B to these Findings)

1.1-350 Employing source term assumptions which the Staff and Licensees agree are conservative, (see Findings 1.1-308), the Licensees' median (50 percent confidence level) frequency estimate for accidents at Indian Point 2 which would cause one or more early fatalities is once in 110,000,000 reactor years, the median frequency for accidents causing 100 or more early fatalities is once in 500,000,000 reactor years, and for 1000 or more early fatalities, the median frequency is

once in 40,000,000,000 reactor years. Comparable values for the 90 percent confidence level are once in 10,000,000 reactor years, once in 50,000,000 reactor years and once in 800,000,000 reactor years. (Licensees 1, p. 29)

1.1-351 Similarly, for accidents at Indian Point 3, the Licensees' median frequency (50 percent confidence level) of one or more early fatalities is once in about 140,000,000 reactor years, the median frequency of 100 or more early fatalities is once in about 700,000,000 reactor years, and the median frequency for 1,000 or more early fatalities is once in about 45,000,000,000 reactor years. Comparable estimates for the 90 percent confidence level are once in about 10,000,000 reactor years, once in about 60,000,000 reactor years, and once in about 1,000,000,000 reactor years. (Licensees 1, p.30)

1.1-352 When pessimistic RSS-type source terms are assumed, the largest number of early fatalities resulting from an accident at either Indian Point 2 or 3 as calculated by the authors of IPPSS would be approximately 7,000 people, the associated probability of which would be about one chance in 25,000,000,000 years. (Licensees 1, p. 26-26a)

1.1-353 The Staff presented a slightly higher estimate of early fatality risk from accidents at Indian Point. Their "best estimate" early fatality risk curve, which employed source terms acknowledged to be overly conservative, (see Finding 1.1-308) and which assumed very long delays in evacuation for earthquake and hurricane initiated accidents, shows, for accidents at Indian Point 2, one or more early fatalities occurring with a frequency of about once in 200,000 years, 100 or more early fatalities occurring about once in 400,000 years, and 1000 or more such fatalities occurring about once in 600,000 years. (Acharya Testimony, Figure III.C.1)

1.1-354 On a similar basis for accidents at Indian Point 3, the Staff's best estimate early fatality curve shows a frequency of about once in about 500,000 reactor years for one or more, once in about 1,000,000 reactor years for 100 or more, and once in about 1,670,000 reactor years for 1,000 or more. (Acharya Testimony, Figure III.C.12)

1.1-355 The largest number of early fatalities resulting from accidents at Indian Point as calculated by the Staff would be about 30,000 people, the associated probability of which would be about one chance in

100,000,000 years. (Rowsome/Blond IV.A, p. 10; Acharya Testimony, Figures III.C.1 and III.C.12) The Staff admitted that this estimate of the largest number of early fatalities would be associated with the simultaneous occurrence of the "largest" release category, the "worst" weather conditions, the wind blowing in the direction of the "highest" population, and a very pessimistic emergency response assumption. (Rowsome/Blond IV.A, p. 10)

(b) Individual Risk

1.1-356 The Licensees estimate the early fatality risk to individuals living within one mile of Indian Point to be 7.1×10^{-9}, or about one in 140,000,000 per reactor year and 6.4×10^{-9}, or about one in 160,000,000 per reactor year, for Units 2 and 3 respectively.* (Paddleford, et al., p. 11; Licensees 1, pp. 31-32)

1.1-357 The Staff concluded that the early fatality risk to individuals from the operation of the Indian Point units is very low. As a function of distance out to 30 miles from the plant, their estimate ranges (on a per

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* See Foonote page 1-169

site basis) from one in 1,000,000 per reactor year to one in 10,000,000,000 per reactor year. (Rowsome/Blond IV.A, p. 11; Acharya Testimony, Figure III.C.23) Additionally, information compiled by the Staff from various risk assessments depicts the risk of early fatality to an individual within one mile of the Indian Point site due to reactor accidents at Indian Point 2 and 3 to be about one in 33,000,000 per reactor year and about one in 1,000,000,000 per reactor year, respectively. (Paddleford, et al., pp. 10-11)

(c) Evaluation of Difference in Risk Estimates

1.1-358 One major reason for the differences in the Licensees' and Staff's estimates of early fatalities can be traced to differences in offsite emergency response assumptions which have an important impact in calculating early fatalities. (Potter 5, p. 6; Acharya Testimony, p. III.C.A-37)

1.1-359 For example, the time available for emergency response is very important to limiting early fatality risk. (Potter 5, p. 5) The Staff analyzed release category RC-C, similar to IPPSS 2RW, using pessimistic assumptions, i.e., the Late Reloc model, (see

Finding 1.1-246) regarding offsite emergency response to earthquake and hurricane-initiated releases. (Acharya Testimony, p. III.C. A-33; Rowsome/Blond IV.B, p. 16) They assumed that for 24 hours after the start of exposure, evacuation or relocation would not take place and, further, that people are out in the open losing benefit of any shielding from buildings and other structures in the region during that time. (Acharya Testimony, pp. III.C. A-10, III.C.A-12, III.C.A-33). They reasoned that emergency response would be degraded by earthquakes and high winds.

1.1-360 The Late Reloc assumptions are comparable to no emergency response for 36 to 48 hours after the initiating event since containment failure and start of exposure is not expected to occur for at least 12 hours, and more likely 24 hours, after the initiating event. (Potter 5, p. 6) For this case, mean early fatalities (conditional on an RC-C release) of 493 and 583 were calculated for Indian Point 2 and 3, respectively. (Acharya Testimony, Table III.C-5)

1.1-361 The authors of IPPSS analyzed evacuation delays probabilistically for all accident scenarios to account for the possibility that emergency response could

be slower than expected for any reason, including adverse weather and road conditions. Thus, delay times as long as 13 hours were used for certain segments of the population. (Licensees 1, p. 121; see Finding 1.1-240)

1.1-362 In the IPPSS, release category 2RW was not a contributor to early fatality risk primarily because with more reasonable emergency response assumptions there was sufficient time for effective evacuation. (Potter 5, p. 5) The Staff also analyzed release category RC-C (approximately equivalent to IPPSS release category 2RW) using evacuation as the selected emergency response (Evac-Reloc model) and reached the same conclusion -- no early fatalities. (Acharya Testimony, Table III.C-5) 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. (Acharya Testimony, Tables III.C-2, III.C-3)

1.1-363 The Evac Reloc evacuation assumptions used by the Staff were even more pessimistic than those in the IPPSS and still 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. (Potter 5, p. 5)

1.1-364 All parties agreed, and the Board finds, that the Staff Late Reloc assumptions, (see Finding 1.1-246), regarding offsite emergency response to earthquake and hurricane initiated releases, are conservative and overstate the early fatality risk. (Potter 5, p. 6; Acharya Testimony, p. III.C.A-33, Rowsome/Blond IV.B, p. 16; Sholly, Tr. 8421, 8423) UCS/NYPIRG witness Sholly characterized the Staff calculation as a bounding and worst case analysis. (Tr. 8421, 8423)

1.1-365 Another reason for the difference in early fatality risk estimates between the Staff and Licensees is the treatment of the release category related to direct seismic failure of the containment building (Staff RC-A, Licensees Z-1Q). The Staff assessment includes contributions to the early fatality risk from an earthquake-initiated collapse of the containment building. (Acharya Testimony, Table III.C-5) The Licensees testified that the Indian Point containment buildings are strong enough to withstand the largest earthquake that could occur in the vicinity (Licensees 1, p. 65; Perla, Tr. 13006; Bley, Tr. 12990; see Finding 1.1-83), and that there is no contribution to the early fatality risk from this release category. (Potter 5, p. 6) At the time of

the hearings the Staff had not yet evaluated the Licensees' testimony that there can be no direct seismic failure of the Indian Point containments but agreed that, if the Licensees' reanalysis was correct, the overall risk would be reduced. (J. Meyer, Tr. 12507-12516)

1.1-366 Elimination of Staff release category RC-C as a contributor to early fatality risk and the elimination altogether of release category RC-A reduces the Staff's estimate of overall risk and the mean early fatality risk by factors of 25 and 6 for Indian Point Units 2 and 3, respectively. (Potter 5, p. 6; Acharya Testimony, Table III.C-5; J. Meyer, Tr. 12509)

1.1-367 The Staff also indicated that the source term they used may have been significantly overestimated. The impact of lesser values of source terms would be substantially lower estimates of health effects, particularly early fatalities. (Acharya Testimony, pp. III.C.A-20 to C.A-21, III.C.A-25) Anticipated reductions in the source terms would reduce the early fatality risk. (See Findings 1.1-302 and 1.1-409) The Licensees testified that a reduction of a factor of 15 in the IPPSS source term would effectively eliminate the early fatality risk, assuming evacuation as modeled in the IPPSS. (Licensees 1, pp. 125-127)

(d) Dominant Contributors

1.1-368 The Staff found that more than 90% of the expected early fatality risk at Indian Point was dominated by accidents initiated by earthquakes and hurricanes (Acharya Testimony, p. III.C.A-32; Rowsome/Blond IV.A, p. 9; Rowsome/Blond IV.B, p. 9), primarily because they pessimistically assumed that neither anticipatory evacuation nor early relocation was feasible for such scenarios. (Rowsome/Blond IV.B, pp. 8, 16) Additionally, they included contributions to early fatality risk from direct seismic failure of the containment buildings. (J. Meyer, Tr. 12507-12516)

1.1-369 Testimony in the hearings was uncontroverted that the Indian Point containments could withstand the largest earthquake which could possibly occur in the vicinity, and that the containments could also withstand internal pressures of 141 psia. (See Findings 1.1-83) This being the case, the dominant contributor to early fatality risk at the Indian Point units is the interfacing systems LOCA, which accounts for more than 97% of the early fatality risk. (Licensees 1, p. 68) This accident, which bypasses containment, is initiated by failure of two

valves which isolate the residual heat removal (RHR) system from the primary coolant system. Since the primary system is usually at much higher pressure, the failure of the valves can result in an overpressurization of the RHR system and loss of primary water. (Licensees 1, p. 33) This accident scenario has a median frequency of occurrence of once in 25,000,000 reactor years. (Paddleford, et al., p. 13) The Staff's point estimate frequency for the interfacing systems LOCA is once in 2,500,000 reactor years. (Israel, et al., pp. 14-15) Although the interfacing systems LOCA is rare, nevertheless this event still dominates the early fatality risk because all other contributors are even lower. (Licensees 1, p. 33)

(e) Distribution of Risk

1.1-370 Licensees estimate that more than 95 percent of the early fatality risk from all core melt accidents, even assuming that there would be no protective measures taken by the public, is confined to within 4 miles of the site. Less than 0.2 percent of the early fatality risk is beyond 10 miles, and less than 0.01 percent is beyond 20 miles. (Licensees 1, pp. 112-113; Potter, Tr. 7642, 7695) The Staff indicated that lethal exposures would probably be limited to a couple of miles

from the reactor. However, for rare adverse meteorological conditions (e.g., downwind precipitation), it would be possible to have very localized high concentration areas that could give lethal exposures at distances out to about 30 miles. (Rowsome/Blond IV.A, p. 9; Acharya Testimony, Figure III.C.50; Rowsome/Blond IV.B, p. 11)

2. Latent Cancer Fatality Risk

(a) Societal Risk

1.1-371 The risk of latent fatalities is very small. Licensees and Staff reached approximately the same conclusion with respect to their estimates of latent cancer fatalities. (Acharya Testimony, p. III.C.A-41, Figure III.C. 58)

1.1-372 The Licensees determined latent cancer fatality societal risk curves for reactor accidents at Indian Point 2 and 3. (Licensees 1, pp. 26-26a (Figures III-1e and III-2e))(Appendices A and B to these Findings). Since latent cancer fatalities begin after some dormancy period and occur over a period of up to 30 years or more, a second scale has been added to the horizontal axis to show the percent increase above the background (i.e., non-nuclear) cancer fatality rate. For purposes of

estimating this rate, it was assumed that all cancer fatality were limited to the 50-mile population. (Licensees 1, p. 27; Rowsome/Blond IV.A, p. 12)

1.1-373 The Licensees' estimate of the median (50 percent confidence level) frequency of accidents occurring at Indian Point 2 causing one or more latent fatalities was about once in 15,000 reactor years, about once in 23,000 reactor years for 100 or more, and about once in 42,000 reactor years for 1,000 or more. The comparable 90 percent confidence level values are about once in 7,300, 9,200 and 13,000 reactor years, respectively. (Licensees 1, pp. 26 (Figure III-1e), 29)

1.1-374 According to the Staff's point estimate latent fatality risk curve for Indian Point 2, the frequency of one or more latent fatalities is about once in 10,000 reactor years, about once in 20,000 reactor years for 100 or more and, about once in 25,000 reactor years for 1,000 or more. (Acharya Testimony, Figure III.C.6)

1.1-375 For Indian Point 3, the Licensees' estimate of the median frequency is once in 58,000 reactor years for one or more latent cancer fatalities, once in 170,000 reactor years for 100 or more, and once in 290,000

reactor years for 1,000 or more. The parallel numbers at the 90 percent confidence level are once in 18,000 reactor years, once in 38,000 reactor years, and once in 61,000 reactor years, respectively. (Licensees 1, pp. 26a (Figure III-2e), 30)

1.1-376 The Staff's point estimate latent fatality risk curve for Indian Point 3 shows the frequency of one or more latent cancer fatality is once in about 10,000 reactor years, the frequency of 100 or more latent cancer fatalities is once in about 33,000 reactor years, and the frequency of 1,000 or more such fatalities is once in about 50,000 reactor years. (Acharya Testimony, Figure III.C-17)

1.1-377 The latent cancers that could result from an accident at Indian Point 2 or 3 would represent only a slight, and probably not statistically significant, increase above the naturally occurring cancer rate. (Licensees 1, p. 27; Rowsome/Blond IV.A, p. 12) The median frequency of accidents at Indian Point 2 and 3 resulting in an increase of 0.01 of one percent or more in the background cancer fatality rate is once in 23,000 and once in 170,000 reactor years, respectively, the median frequencies of exceeding these rates by 0.1 percent are

once in 50,000 and once in 330,000 reactor years, respectively; and the median frequencies of exceeding them by one percent are once in 10,000,000 and 100,000,000 reactor years, respectively. (Licensees 1, pp. 26 (Figure III-1e), 26a (Figure III-2e), 29, 30)

(b) Individual Risk

1.1-378 The Licensees' estimate of the average latent cancer fatality risk to an individual within one mile of Indian Point is 6.2×10^{-8}, or about one in 16,000,000 per reactor year, and 1.6×10^{-8}, or about one 62,000,000 per reactor year, for accidents at Indian Point 2 and 3 respectively.* (Licensees 1, pp. 31, 32) The Staff presented estimates of the individual risk of delayed (latent) cancer fatalities on a per site basis as a function of distance. Their estimate of the individual risk is low, ranging from about one in 1,200,000 per reactor year to about one in 300,000,000 per reactor year. The latent cancer fatality risk to an individual within one mile of the site is about one in 1,700,000 per reactor years (Acharya Testimony, Figure III.C.26). The average,

* Licensees' Testimony (pages 31-32) incorrectly converted the value of 6.1×10^{-8}; the correct values are about 16,000,00 and 62,000,000 respectively.

expected, number of latent fatalities per reactor year within 50 miles of Indian Point, as calculated by the Licensees, is 0.21 and 0.05 resulting from accidents at Indian Point 2 and 3 respectively. (Licensees 1, p. 39) Staff, combining the risk from Indian Point Units 2 and 3, reported 0.246 total expected cancer fatalities per reactor year. (Rowsome/Blond 5, pp. B-15, B-16)

(c) Dominant Contributors

1.1-379 The dominant contributor to latent fatality risk at the Indian Point plants is a radioactive release associated with gradual overpressurization of containment (release category 2RW). (IPPSS Amendment 1 at 8.1-4; Licensees 1, pp. 33-34; Rowsome/Blond IV.A, p. 8, Figure IV.A.2) The median frequency of containment overpressurization failures at Indian Point 2 and 3 is once in 30,000 and once 170,000 reactor years, respectively. (Licensees 1, p. 24; see IPPSS Amendment 1, Table 8.1-1, p. 8.1-5) Staff's calculated frequency for this release is about once in 29,000 and once in 57,000 reactor years for Indian Point 2 and 3, respectively. (Meyer/Pratt Testimony, pp. III.B-26, Figures III.B-9, 10)

1.1-380 External initiating events are the major contributors to latent fatality risk from accidents at

both Indian Point plants. At Indian Point 2, the largest of these contributors is severe wind (hurricane) well beyond the highest level ever recorded in the Indian Point area. Such a hurricane was assumed to cause loss of AC power which, assuming no recovery, would lead to a late melt and subsequent containment overpressurization. The mean frequency of this scenario is about once in 37,000 reactor years. (Licensees 1, pp. 33-34) With the hurricane watch in effect, this frequency is substantially lower. (Rowsome/Blond III.A.4, pp. 4, 6) The second largest contributing scenario is initiated by a tornado which would cause simultaneous loss of onsite and offsite power and other safety related equipment. The mean frequency of this scenario is about once in 63,000 reactor years. The next largest contributor is a seismically induced complete loss of power scenario which, assuming no recovery, would lead to a late melt and delayed overpressurization of containment. The mean frequency of this scenario is about once in 145,000 reactor years. (Licensees 1, pp. 33-34)

1.1-381 Fires originating in either the switchgear or the cable spreading room are the main contributor to latent fatality risk from accidents at Indian Point 3. (IPPSS Amendment 1, Table 7.3.2A-14, p. 7.3-123A-2;

Licensees 1, p. 34; Rowsome/Blond IV.A, p. 7, Figure IV.A.2) These fires could lead to a loss of AC power and eventual containment overpressurization. The mean frequency of fires leading to latent fatalities is once in 140,000 reactor years. (Licensees 1, p. 34) The next largest contributor to latent fatalities is a large seismic event causing loss of control power or loss of AC power. This accident scenario has a mean frequency of once in 400,000 reactor years. (Licensees 1, p. 34; Rowsome/Blond IV.A., Figure IV.A.2)

3. Other Health Indices

1.1-382 With regard to other health indices, the Licensees presented societal risk curves for non-fatal radiation illnesses and thyroid cancers resulting from accidents at Indian Point 2 and 3, respectively. (Licensees 1, p. 26 (Figures III-1b, d), 26a (Figures III-2b, d)) These figures are attached to these Findings as Appendices A and B.

1.1-383 The median frequency of accidents at Indian Point 2 causing one or more non-fatal radiation illnesses is once in 2,000,000 reactor years; for 100 or more such illnesses, once in 20,000,000 reactor years; and

once in 750,000,000 reactor years for 1,000 or more illnesses. (Licensees 1, pp. 26, 29) The Staff's point estimate early injury (radiation illness) risk curve shows the frequency of one or more radiation illnesses to be about once in 25,000 reactor years, the frequency of 100 or more to be about once in 33,000 reactor years, and the frequency of 1,000 or more to be about once in 50,000 reactor years. (Acharya Testimony, Figure III.C.31)

1.1-384 The median frequency of one or more non-fatal radiation illnesses at Indian Point Unit 3 is once in 6,400,000 reactor years, the median frequency of 100 or more illnesses is once in 45,000,000 reactor years, and the median frequency of 1,000 or more such illnesses is once in 950,000,000 reactor years. (Licensees 1, pp. 26a, 30) According to the Staff's point estimate early injury (radiation illness) risk curve, the frequency of one or more radiation illnesses is once in about 50,000 reactor years, the frequency of 100 more radiation illnesses is once in about 100,000 reactor years, and the frequency of 1,000 or more radiation illnesses is once in about 160,000 reactor years. (Acharya Testimony, Figure III.C.42)

1.1-385 The Licensees estimated the average risk of radiation illness to an individual within one mile of

the site from accidents at Indian Point 2 and 3 to be 7.4 x 10^{-9}, or about one in 14,000,000 per reactor year and 2.1 x 10^{-8}, or about one in 48,000,000 per reactor year, respectively.* (Licensees 1, pp. 31 32) The Staff presented the individual risk of early injury (radiation illness) on a per site basis as a function of distance. Their risk estimate is low, ranging from about one in 1,100,000 per reactor year to 1,250,000,000 per reactor year. The risk of early injury caused by Indian Point to an individual one mile from the site is about one in 1,100,000 per reactor year on a per site basis. (Acharya Testimony, Figure III.C.25)

1.1-386 The Licensees estimate of the median frequency of accidents at Indian Point 2 causing one or more thyroid cancers is about once in 8,500 reactor years; about once in 54,000 reactor years for 1,000 or more thyroid cancers. (Licensees 1, pp. 26, 29) Thyroid cancers are curable in approximately 90 percent of the

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* Licensees' Testimony (pages 31-32) incorrectly converted the value of 7.4 x 10^{-8} and 2.1 x 10^{-8}; the correct values are about 14,000,000 and 48,000,000 respectively.

cases. (Id., p. 18). The Staff presented point estimate risk curves only for those thyroid cancers which are fatal, while Licensees computed consequences for all thyroid cancers. (Licensees 1, p. 18; Acharya Testimony, p. III.C.A-41) Thus, to facilitate comparisons, Staff's estimated consequences must be multiplied by 10 since only one thyroid cancer out of ten is fatal. The Staff's point estimate thyroid cancer fatality risk curve indicated a frequency of once in about 10,000 reactor years for 10 or more thyroid cancers (one fatality) and about once in 30,000 reactor years for 1,000 or more thyroid cancers (100 fatalities). (Acharya Testi- mony, Figure III.C.35)

1.1-387 The median frequency of one or more thyroid cancers from accidents at Indian Point 3 is once in 25,000 reactor years, of 100 or more such cancers it is once in 180,000 reactor years, and of 1,000 or more it is once in 350,000 reactor years. (Licensees 1, pp. 26a, 30) The thyroid cancer fatality risk curve presented by Staff showed a frequency of about once in 10,000 reactor years for 10 or more thyroid cancers (one fatality) and a frequency of about once in 20,000 reactor years for 1,000 or more thyroid cancers (100 fatalities). (Acharya Testimony, Figure III.C.46)

1.1-388 The Licensees demonstrated that the conditional probability of an individual near the site boundary receiving a radiation dose in excess of 200 rem to the whole body, given a core melt at Indian Point 2 and 3, is less than 0.3 and 0.085, respectively. Ten miles from the plant, the conditional probability is about .003 and .001, respectively, for Indian Point 2 and 3. This analysis assumed that the public would take no protective action (evacuation or shelter) following an accident for 24 hours. (Licensees 1, pp. 112, 114-115)

4. Offsite Property Damage

1.1-389 With regard to offsite property damage, the Licensees constructed a family of risk curves expressed in 1982 dollars to display the property damage that may result from reactor accidents at Indian Point 2 and 3. (Licensees 1, pp. 26 (Figure III-1f), 26a (Figure III-2f)) These figures are attached to these Findings as Appendices A and B.

1.1-390 The median frequency of any offsite property damage caused by an accident at Indian Point 2 and 3 is once in 6,700 and once in 17,000 reactor years respectively. (Licensees 1, pp. 26-26a, 28) The authors

of IPPSS performed separate computer runs for each release category, (IPPSS, p. 0-62), including benign release category 8B, which occurs with each postulated core melt accident but is equal only to normal levels of containment leakage. (IPPSS, Table 5.4.1, p. 5.4-6) Because each computer run assumed evacuation, the costs of which are included in estimates of property damage, the median frequency of any offsite property damage approximately equals the core melt frequency. (IPPSS Amendment 1, Figure 8.1-5, at p. 8.1-7)

1.1-391 Staff presented a point estimate property damage curve in 1981 dollars which shows the frequency of any offsite property damage from accidents at Indian Point 2 and 3 to be once in 3,300 and once in 10,000 reactor years respectively. (Acharya Testimony, Figures III.C.10 and III.C.21)

1.1-392 Interdiction of those areas where reasonable decontamination procedures could not reduce the contamination to acceptable levels is the major contributor to property damage estimates. Costs associated with decontamination and relocation expenses would also be significant. The other costs, associated with agricultural losses and evacuation, would not be significant in comparison to the total costs. (Rowsome/Blond IV.A, p. 14)

1.1-393 The Staff's estimate of the annual average property damage risk for the two unit site in 1981 dollars amounts to $450,000 per year. (Acharya Testimony, pp. III. C-8, 9) Although a realistic estimate of the background economic loss rate within 50 miles of the site has not been developed, the Staff concluded that a wide variety of accidental hazards would pose economic loss rates well in excess of this rate. (Rowsome V, p. 3)

5. Risk In Perspective

1.1-394 Licensees presented the testimony of Dr. Bernard L. Cohen with regard to the level of risk posed by the continued operation of Indian Point as compared with other everyday risks in society. (B. Cohen Testimony)

1.1-395 The operation of Indian Point is only one of may sources of risk in society. However, it is important to be concerned about significant risks, rather than being preoccupied with trivial risks. Some members of the public misunderstand the risk of Indian Point because most people do not understand and quantify the other risks that we routinely face. Many people think and

act as though life should be largely free of risk. (B. Cohen Testimony, p. 2; B. Cohen, Tr. 14429)

1.1-396 Everything we do involves risk. This does not mean that we should not try to minimize our risks. However, we cannot eliminate risk by simply avoiding those that we choose to think about. The logical procedure for both personal and societal decision-making in order to minimize risks is to quantify all risks and then to accept smaller risks in preference to larger ones. (B. Cohen Testimony, pp. 2-3)

1.1-397 There are several ways of expressing quantified risks, one of which is Loss of Life Expectancy (LLE), which computes the average amount by which one's life is shortened by the risk under consideration. (B. Cohen Testimony, pp. 3, 28-30)

1.1-398 For example, smoking one pack of cigarettes per day (male/female - average) causes a loss of life expectancy of 1,600 days; being fifteen pounds overweight causes a 450 day loss; all accidents cause a 435 day loss; motor vehicle accidents cause a 200 day loss; risk of drowning causes a 40 day loss; exceeding speed limits between 55-65 mph causes a 40 day loss; diet

drinks cause a 2 day loss; airplane crashes cause a 1 day loss; all electric power causes a .03 day loss; and spending a lifetime near Indian Point causes a .006 day (or 9 minutes) loss. Other risks include working in manufacturing and service industries, which has an LLE of 45 days, use of coffee, which has a 6 day LLE and birth control pills, which give their users an LLE of 5 days. (B. Cohen Testimony, pp. 3-10, 28-30)

1.1-399 Many people believe that large catastrophes pose important threats to society. This belief is unsupported by LLE figures for hurricanes and tornadoes, which result in an LLE of 1 day. Major fires and explosions cause an LLE of .7 days and massive chemical releases have an LLE of .1 days. The cancers from a hypothetical severe meltdown accident at Indian Point would increase the cancer rate of those exposed by .02 percent whereas the national average cancer rate is approximately 20 percent. (B. Cohen Testimony, pp. 10-11)

1.1-400 The Board has received calculations of the risk of Indian Point which were prepared by the Licensees, NRC and UCS. Licensees have calculated that the mortality risk of residing near Indian Point is approximately 10^{-8}/year. (Licensees 1, p. 27) This is

equivalent to an LLE of .006 days (9 minutes). (B. Cohen Testimony, p. 12) The Reactor Safety Study (WASH-1400) estimates that the risk of nuclear reactor accidents would reduce life expectancy by .012 days, or 18 minutes, and the Union of Concerned Scientists (UCS) assessed that risk at 1.5 days. (B. Cohen Testimony, pp. 12, 28-29) However, the Three Mile Island incident gave the average Harrisburg area resident an LLE of only 1.5 minutes (.001 days). (B. Cohen Testimony, pp. 3-10, 28-30) Thus, nuclear accidents are thousands of times less dangerous than other risks in society. (B. Cohen Testimony, p. 12)

1.1-401 Indian Point is opposed largely because of the risk it is thought to pose. However, the best way to understand risk is to compare it with other types of familiar risks. Comparisons show that the risks of nuclear plants in general, and Indian Point in particular, are small when compared with other every day risks to which the public is exposed. (B. Cohen Testimony, pp. 13-14)

1.1-402 The Staff also did an evaluation of how the risks posed by Indian Point compared against the background of competing non-nuclear risks, including rare, high consequence risks. (Rowsome V, pp. 2-6)

1.1-403 This evaluation showed that severe accidents at the Indian Point plants represent roughly 2.5 parts per million of the background early fatality risk and roughly 11 parts per million of the background latent fatality risk within a 50-mile radius of the plants. With regard to offsite property damage, the Staff found that other accidental hazards pose economic loss rates well in excess of those posed by the two unit site. (Rowsome V, p. 3)

1.1-404 The Staff concluded that the annual average risks posed by severe reactor accidents at Indian Point Units 2 and 3 would not loom large against the background of competing risks, even if the Staff's Indian Point risk estimates were substantially underestimated. (Rowsome V, p. 4)

1.1-405 With regard to rare high consequence risks, the Staff concluded that among rare man-caused accidents having early fatalities in the hundreds or more, the contribution posed by severe reactor accidents at Indian Point is quite small. The Staff similarly concluded that the increment in the cancer rate each year resulting from an accident at Indian Point would not loom large

against the non-nuclear background cancer fatality rate. (Rowsome V, p. 5)

1.1-406 The Licensees and Staff also testified that the public health risks posed by the Indian Point plants melt the Commission's public health safety goals. (See Findings 5.0-6, 5.0-7)

O. ULTIMATE FINDINGS

1.1-407 The Board finds that both Licensees and Staff performed thorough PRAs. IPPSS, performed by Licensees, is a major advance in PRA methodology, including a state-of-the-art quantification of uncertainty, a pioneering analysis of external initiating events, a sophisticated data base including plant-specific data, and a realistic consequence model. The Board finds that the risk curves presented in Appendices A and B to these Findings constitute reasonable estimates of risk associated with the operation of the Indian Point plants. Further the Board finds that, as both the Staff and Licensee risk assessments concluded, and as Appendices A and B show, the risk from Indian Point Units 2 and 3 for every health effect studied is extremely low.

1.1-408 The Board finds that the principal reason for the extremely low risk at Indian Point is the strength of the containments. First, based on the evidence and cross-examination, the Board finds that any uncertainties with regard to the strength of the Indian Point containments have to do exclusively with the calculation of the ultimate failure pressure of the containments and not with the pressure at which onset of yielding first occurs. For purposes of both the Staff's and Licensees' analyses, the onset yielding has been conservatively assumed to be the actual failure pressure of the containment. Since UCS/NYPIRG has done no independent analysis of containment failure pressure, and since both the Staff and Licensee analyses incorporate sufficient conservatism, the Board finds that no breach of the Indian Point containments would occur at pressures below 141 psia. Second, the uncontroverted evidence indicates that the containments can withstand any seismic accelerations which could affect the site, thereby making direct failure of the containment impossible.

1.1-409 Based upon the evidence and cross-examination of the Staff and Licensees, the Board finds that the source terms employed in the Staff analysis and in the IPPSS are conservative and result in overestimates of the

already low risk from operation of the Indian Point facilities. While the precise magnitude of the anticipated reduction in source terms is not clear at this time, the evidence tends to support the likelihood that at least a factor of 10 reduction will be forthcoming. Such a reduction would dramatically reduce the consequences of even the most severe reactor accidents.

1.1-410 The Board has heard testimony regarding consequence estimates made by Palenik/Beyea and Pisello/Piccioni on behalf of several intervening parties. However, the Board finds that, despite the direction of the Commission to the parties in this proceeding to consider probability and consequences equally, no intervening party presented any probability estimates for the consequence calculations made by these witnesses. Further, the evidence indicates that many of the assumptions made by Palenik/Beyea and Pisello/Piccioni are either clearly unrealistic or involve very low frequency combinations of meteorological conditions. The failure of Palenik/Beyea and Pisello/Piccioni to present probabilistic assessments coupled with the unrealistic assumptions employed in their respective analyses leads the Board to conclude that their testimony is of little probative value.

III. BOARD QUESTION 1.2

1.2-1 Board Question 1.2 reads as follows:

> 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?

1.2-2 The Licensees and the NRC Staff filed direct testimony with regard to this Board question. No direct testimony was filed by any of the intervening parties or interested states.

1.2-3 NUREG/CR-2497 was prepared with the objective of identifying those events that could have contributed directly to the occurrence of core damage or core melt accidents between 1969 and 1979. Science Applications, Inc., working on contract to Oak Ridge National Laboratory, compiled this list of precursors for potential severe core damage accidents by sifting through approximately 19,400 Licensee Event Reports (LERs) with the objective of calculating the frequency of severe core

damage accidents in commercial nuclear power plants.

(Rowsome 1.2, p. 3; Kaplan 1.2, pp. 1-2)

1.2-4 LERs included in the report were selected against six screening criteria which included:

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 off-site power and any less frequent off-normal initiating events or accidents;

5. any event or operating condition that was or proceeded differently from the plant design basis; and

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

(Rowsome 1.2, p. 4)

1.2-5 Of the events selected to be included in the analysis, based on the screening criteria, one of the events was actually a core damage accident, namely the accident at Three Mile Island. The period of the study, the decade of the seventies, included 432 reactor years of

experience from domestic reactors. Therefore, the authors of the study arrived at an estimate of the industry average core melt frequency for the decade of the seventies by dividing one (the number of severe core damage events) by 432 (number of reactor operating years of experience during the period) and then adding to the TMI event other precursors, weighted by an estimate of the conditional probability that each precursor event might have led to a severe core damage accident. (Rowsome 1.2, pp. 5-6; Kaplan, 1.2, p. 2) In essence, the authors of the NUREG considered that the precursors other than the TMI accident together were the equivalent of about one more severe core damage accident. (Kaplan, 1.2, p. 2) By this weighting process, the industry average core melt frequency for the decade of the seventies was determined to be between 1.7×10^{-3} and 4.5×10^{-3}. (Kaplan 1.2, p. 2; Rowsome, Tr. 8315)

1.2-6 The manner in which the NUREG concluded that near misses were the equivalent of an actual core-melt has been subject to severe criticism. Such conclusion is illogical. Severe criticism has also been made, from an engineering modeling standpoint, of the manner in which near misses were accounted for in the event tree work in the study. (Kaplan, Tr. 8266-8268) Further, a

complete treatment of core melt frequency should be done with full probability curves rather than single number "point" estimates (Kaplan 1.2, p. 7), since full probability curves give more information than point estimates. (Blond, Tr. 8682) Regarding this near miss contribution to severe core damage frequency, a review of the study by the Institute of Nuclear Power Operation (INPO) concluded that the calculation of this contribution is about 30 times too high for the plants as they were at the time of the events. INPO concluded that the NUREG frequencies, on the basis of the treatment of near misses, were not appropriate for estimation of either past or future performance. (Kaplan 1.2, p. 3)

1.2-7 The core melt frequency set forth in NUREC/CR-2497 was calculated without any effort to distinguish between different plants or types of plants. (Kaplan 1.2, p. 1) One thing that has become clear through post Reactor Safety Study (RSS) probabilistic risk assessments is that the frequency of core melt can be very different from plant to plant, because of design differences. Even within the same plant, the core melt frequency can change when design modifications are made. This was observed in the development of the IPPSS when such modifications were made to the Indian Point facilities. (Id., p. 6)

1.2-8 Eighty-two percent of the industry average core damage frequency estimated in the precursor study originated from the three closest calls, the accident at TMI 2, the Browns Ferry fire, and the instrument power supply failure at Rancho Seco. (Rowsome 1.2, p. 6; Rowsome, Tr. 8331) These events were dependent to a great extent on the specifics of the design of each plant. None of them are Westinghouse plants. (Rowsome, Tr. 8332)

1.2-9 With respect to the accident at Three Mile Island, there are fundamental differences in the designs of the Indian Point and TMI plants which make it vastly less likely that scenarios of the TMI-type could occur at Indian Point. These differences include the use at Indian Point of drum-type steam generators with greater heat capacity (in the form of secondary coolant inventory), greater heat capacity of the primary coolant system (in the form of primary coolant inventory), and a reactor trip signal that would respond immediately to a loss of feedwater condition. Hence, at Indian Point, there would be no immediate primary system pressure rise in response to a loss of feedwater. (Kaplan 1.2, p. 5)

1.2-10 Plants with Westinghouse reactors, such as Indian Point, very rarely open their pilot-operated relief valve 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 these valves during feedwater transients. (Rowsome 1.2, pp. 10-11) At Indian Point, even if the auxiliary feedwater were delayed as much as twenty to thirty minutes, the primary system would not experience a pressure transient, and the relief valve would not lift; therefore, it would not stick and could not lead to a loss of coolant. Even if auxiliary feed were delayed longer than twenty to thirty minutes, and even if the relief valve were to lift and stick open, at Indian Point there is direct monitoring of the position of this valve so that the operators would be aware of its condition. They could then act to block the valve and/or inject coolant, thus bringing the plant to a stable condition with no core damage. The probability of achieving this condition of plant stability is greatly enhanced at Indian Point by the elevated steam generator design. (Kaplan 1.2, pp. 5-6) 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 Babcock and Wilcox plants. (Rowsome, 1.2, p. 11)

1.2-11 With regard to the operator errors which contributed to the TMI accident, no probabilistic risk assessment, including the IPPSS and Staff analyses has yet done a thorough job of screening accident scenarios for cases in which the operators might be misled into a faulty diagnosis of the basic scenario. However, it can be said that since the TMI accident, operators are far better trained to recognize inadequate core cooling and new instruments have been provided to aid in that diagnosis, so that it is far less likely today that operators would turn off needed safety systems than it was prior to TMI. (Rowsome 1.2, p. 11) Operator error such as occurred at TMI is quite unlikely today because the accident at TMI raised consciousness levels of all nuclear plant operators. Procedures are drafted in quite a different way today than prior to TMI and operator training is conducted differently. Procedures and training now are more symptom-oriented and focused to a greater extent on diagnostic problems related to serious accidents. Operators are now very well trained to recognize the unmistakable symptoms of severe core damage, such as very high radiation levels in containment. Severe accident scenarios would almost certainly be nipped in the bud before core melt down, as was the accident at TMI. (Rowsome 1.2, pp. 11-12; Rowsome, Tr. 8340-8341)

1.2-12 Because of the above considerations, the frequency of occurrence of TMI-type accidents at Indian Point is very low. However, even if the Indian Point units were subject to a TMI-type severe core damage or even a full meltdown accident, such an accident 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 perform its job of mitigating the accident. (Rowsome 1.2, pp. 10, 12) Even if such accidents were to have a high frequency at Indian Point, they would contribute very little to off-site radiological risk. (Id., p. 10)

1.2-13 With regard to the second most significant precursor event identified in NUREG/CR-2497, the Browns Ferry fire, interim actions taken on the basis of that event have reduced the fire risk at the Indian Point plants. (Rowsome 1.2, p. 16) Fire was considered as an initiating event in the IPPSS. (See Findings 1.1-94 to 1.1-107; Rowsome 1.2, p. 12) In arriving at core melt frequencies for fire initiated accidents, the frequencies

of fires were derived in the IPPSS from evidence collected from all U.S. nuclear power generating stations. It is debatable whether actual fire occurrences such as Browns Ferry should have been part of the data base because modifications have been implemented as a result of that fire. Nevertheless, all these fires, including Browns Ferry, were included in the data base. (IPPSS, p. 7.3-3) In addition, the current regulatory programs to enforce the new fire protection rules, Appendix R to 10 C.F.R. Part 50, will still further reduce the susceptibility of the Indian Point facilities to in-plant fires. (Rowsome 1.2, p. 16)

1.2-14 The Rancho Seco incident was ranked in NUREG/CR-2497 as the third most significant precursor event. The concern raised by this incident lies in the fact that one power supply fault could cause loss of main feedwater, disable or potentially disable the auto start of the emergency feedwater system, and blind the operator to the need to turn on either 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 non-safety-related instrumentation, and have four redundant trains so that no one power supply fault could be so disruptive or dangerous. (Rowsome 1.2, p. 13)

1.2-15 As with the TMI accident, the Indian Point units were less susceptible than the Rancho Seco facility to the type of event which occurred there, even before the lessons of that occurrence were learned, and follow-up changes in procedures and operator training make that type of incident far less likely than it was previously. (Rowsome 1.2, p. 16)

1.2-16 The Rancho Seco incident did create the inference that probabilistic risk assessments should analyze those accidents that might be precipitated by a loss of an auxiliary system such as an instrument power supply. Such analyses have been performed both in the IPPSS and in the Staff's review of that document. (Rowsome 1.2, p. 13)

1.2-17 The remaining eighteen percent of the industry average core melt frequency estimated in the precursor study, other than the TMI, Browns Ferry and Rancho Seco incidents, originated almost entirely in loss-of-main-feedwater and loss-of-off-site power precursors. No one of them stands out as particularly important. Indian Point Units 2 and 3 are each less susceptible to core damage from such scenarios than the

precursor study suggests. 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. There are also alternative sources of ac power available to the Indian Point units for loss-of-off-site power incidents not shared by other plants or considered in the precursor study. In particular, the NUREG/CR-2497 analysis is unduly pessimistic in this regard when applied to Indian Point because it did not credit the possibility that one of the three gas turbines at the site or the Buchanan substation could be started and employed to replace off-site power. (Rowsome 1.2, pp. 14, 16-17)

1.2-18 Based on the above considerations, it is quite clear that both Indian Point Units 2 and 3 are less likely to suffer a severe core damage accident of the kinds emerging as significant in the precursor study than that study suggested was true of the average plant in the 1969-79 period. The industry average core damage frequency calculated for the decade of the seventies in NUREG/CR-2497 is therefore an unduly pessimistic predictor

of the frequency of such reactor accidents at either Indian Point Unit today. (Rowsome 1.2, pp. 16, 17; Rowsome, Tr. 8278) In addition, the precursor report does not 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. (Rowsome 1.2, pp. 16, 17; Rowsome, Tr. 8278)

1.2-19 The data reported in NUREG/CR-2497, and the precursor inciderts analyzed there were well known to the IPPSS analysts at the time of preparation of their study. These events have been taken into consideration and included in the data base used in the IPPSS modeling and the frequency calculations, along with data from the IPPSS team's own review of the licensee event reports and, more importantly, along with the specific operating data from the Indian Point plants. The publication of the precursor study thus provided no new information to the IPPSS analysts. It did, however, focus attention on the role of human errors of commission in the accidents in the decade of the seventies. The IPPSS scenarios were therefore reexamined by the IPPSS authors from the human error standpoint and from the standpoint of implementation of

lessons learned from the precursor events. After this reexamination, the IPPSS analysts confirmed that the probability curves for the IPPSS scenario frequencies accurately expressed the current state of knowledge of those frequencies, and that none of the presentations of risk set forth in IPPSS or in the Licensees' risk testimony required a change on the basis of information contained in the precursor study. (Kaplan 1.2, pp. 4-5; Kaplan, Tr. 8245-8246) There was no testimony by any party to the contrary. Indeed, a complete treatment of core melt frequency should consider full probability curves, as was done in the IPPSS, rather than single number point estimates, such as those employed in the precursor study. (Kaplan 1.2, p. 7)

1.2-20 The NUREG range of frequency may be high by as much as factor of 10, and low by as much as a factor of three. (Rowsome, Tr. 8282-8283) The core melt probabilities estimated in IPPSS are approximately one order of magnitude lower than those estimated in the NUREG. This difference does not invalidate IPPSS, nor does IPPSS invalidate the NUREG. (Rowsome, Tr. 8308-8309) One cannot contradict the other because they measure different frequencies: the NUREG estimates an industry average of SCD (i.e., core melt) while IPPSS

estimates a corresponding frequency of a particular plant, Indian Point. (Rowsome, Tr. 8313-8314) It cannot be concluded from the NUREG that the uncertainty range reported in IPPSS is too small. (Rowsome, Tr. 8309-8310) Notwithstanding the different approaches, the NUREG results neither confirm nor refute those reported in IPPSS. (Rowsome 1.2, p. 2)

1.2-21 The importance of the precursor study for the Indian Point plants is further diminished by the finding of both the Licensees' and Staff's analyses that, for Indian Point, core melt frequency itself is a poor indicator of public health risk in the vicinity of these facilities. (Kaplan 1.2, p. 7; see Findings 1.1-127)

A. ULTIMATE FINDING

1.2-22 The Board finds that the results reported in NUREG/CR-2497 have no bearing on the reliability of the IPPSS results. The NUREG's data and the incidents analyzed in it were included in the IPPSS along with specific operating data from Indian Point. Indian Point is less likely than the average plant to experience the precursors weighed most heavily in the NUREG. Further, because the NUREG calculated an industry-average frequency

of SCD accidents in the 1970s, it is unduly pessimistic with regard to the frequency of similar accident sequences at Indian Point.

IV. BOARD QUESTION 1.3

1.3-1 Board Question 1.3 reads as follows:

> What are the probabilities associated with the consequences presented in the testimony of Dr. Beyea and Mr. Palenik?

1.3-2 The parties were invited to address this Board question because, despite the direction from the Commission to equally consider probability and consequences, Palenik/Beyea did not calculate the probability of the PWR-2 release which they postulated for the Indian Point plants. They testified that the probability is so uncertain that it is misleading to state a central estimate and that there was not sufficient experience with reactors to allow a reliable probability estimate. (Palenik/Beyea Testimony, pp. 9, 25, 67-70) Further, Dr. Beyea testified that one must look at the entire risk spectrum, not simply point values, and that the risk spectrum could not be captured with a single number. (Beyea, Tr. 3201) In determining the probabilities of the various accident scenarios analyzed in the IPPSS, the

Licensees have recognized the uncertainties to which Dr. Beyea referred and have identified, quantified, displayed and propagated those uncertainties throughout the IPPSS study. (See Finding 1.1-42) As Dr. Beyea has suggested, Licensees have considered the entire risk spectrum in arriving at the probability of each core melt accident considered in the IPPSS. (See Findings 1.1-29 to 1-30)

1.3-3 The Licensees and the NRC Staff presented direct testimony regarding this board question. None of the intervening parties, including those who had sponsored the Palenik/Beyea testimony, offered direct testimony associating the consequences predicted by Palenik and Beyea with any probability.

1.3-4 Palenik/Beyea presented an estimate of large numbers of cancer fatalities and thyroid nodules resulting from early phase exposure in the New York City area to a release similar to the Reactor Safety Study's release category PWR-2. (Potter 1.3, p. 1; Blond 1.3, p. 2)

1.3-5 A PWR-2 release is more or less equivalent to the Staff's release category A, which is the worst (has the largest source term) release category analyzed for the

Indian Point Units. The probability associated with release category A is 7×10^{-7} and 3.5×10^{-8} per year for Units 2 and 3, respectively. (Blond 1.3, p. 2) In the IPPSS nomenclature, the release categories which dominate the cancer fatality risk are 2 and 2RW. Other release categories are too low either in release quantity or release frequency to contribute to the risk of radiological conditions postulated by Palenik/Beyea for the New York City area. (Potter 1.3, p. 1)

1.3-6 In addition to Palenik/Beyea's choice of a WASH-1400 release category, which does not really represent the Indian Point reactors, there are other difficulties associated with assigning a probability to the Palenik/Beyea consequence estimates. First, the Palenik/Beyea testimony presents consequences for two time invariant weather conditions and two directions. This represents four discrete results. The Staff results developed consequences for a spectrum of realistic weather conditions and do not necessarily even duplicate the time invariant unrealistic conditions used in the Palenik/Beyea analysis. (Blond 1.3, p. 2) Although Palenik/Beyea's choice, for the purpose of meteorological assumptions, of the Pasquill Stability Class D is in the middle of the spectrum of stability classes A - F, such a meteorological

assumption by Dr. Beyea cannot be viewed as a credible effort to be somewhere in the middle of the range of health effects, as he claims his consequences estimates are. (Potter, Tr. 8356-8357) The radiological conditions estimated using Palenik/Beyea's meteorological assumptions are not average, but on the severe side of the distribution. (Potter, Tr. 8358)

1.3-7 Some of the other assumptions used by Palenik/Beyea to perform their calculations are significantly different than the assumptions used by both the Licensees and the Staff in their analyses. These include parameters such as emergency response estimates, distance truncation, health effects conversion factors, exposure time, and shielding factor. (Blond 1.3, p. 3; Potter 1.3, pp. 2-3)

1.3-8 Both the Licensees and the Staff found the validity of Dr. Beyea assumptions questionable. (Potter 1.3, pp. 2-3; Blond 1.3, p. 3; Blond, Tr. 8376) There is no aspect of the Palenik/Beyea analysis that understated the risk, but several ways in which it was overstated. (Potter, Tr. 8353-8354)

1.3-9 In particular, Palenik/Beyea's selection of a dose to health effects conversion factor of 500 cancer fatalities per million man-rem was the upper bound of the range of uncertainty expressed in a dissenting opinion to the 1980 BEIR Report, and is about a factor of three to five times higher than the mid-range values supported by 21 of the 23 committee members. (Potter 1.3, p. 2; see Findings 1.1-320)

1.3-10 Radioactive materials deposited on the ground are large contributors to dose. The dose from ground deposition is highly sensitive to combined assumptions of exposure time and shielding factor (the dose reduction afforded by structural material). The shielding factor assumed by Palenik/Beyea, about equal to that assumed for normal activities in WASH 1400, would be appropriate for a person who spends much of his time on the ground floor of a one-story or two-story house as well as a substantial amount of time outside unshielded. These conditions do not apply to the New York City area where the shielding afforded by larger buildings would be greater. Further, it is likely that doses would be limited by restrictions placed on outdoor time, and other actions such as decontamination or relocation. (Potter 1.3, pp. 2-3)

1.3-11 Given the methodological problems and overly conservative assumptions identified in Findings 1.3-6 through 1.3-10, the Staff testified that any frequency it could assign to the Palenik/Beyea consequence estimates would have little meaning. However, the Staff arrived at a probability estimate for the Palenik/Beyea consequences by comparing the range of delayed cancer deaths predicted by Palenik/Beyea with the complementary cumulative distri- bution functions in the Staff's consequence testimony. Using those curves, the Palenik/Beyea range of delayed cancer fatalities corresponds to a probability range from about 10^{-5} to 10^{-9} for Unit 2, and 5×10^{-6} to below 5×10^{-9} for Unit 3. (Blond 1.3, p. 3; Acharya Testimony, Figures III.C.6, III.C.17)

1.3-12 In estimating the probabilities which were reflected in their testimony, Licensees did not limit their assessment to the specific conditions postulated by Palenik/Beyea, but analyzed all scenarios that could produce consequences which were equivalent to or greater than those calculated by Palenik/Beyea. (Potter, Tr. 8354-8356)

1.3-13 Licensees arrived at an estimate of the probability for the Palenik/Beyea consequence calculations by multiplying the mean frequencies for the 2 and 2RW type release categories by the probability of occurrence of the meteorological conditions required to distribute the material over the New York City area in the way characterized by Palenik/Beyea as "average", and adding contributions from both Indian Point Units 2 and 3. The Licensees' analysis resulted in a total frequency of occurrence for the doses modeled by Palenik/Beyea of 5.5×10^{-6} per reactor year, or once in 180,000 years for the "average" scenario. (Potter 1.3, p. 1) This frequency for the "average" scenario corresponds to the low end of the range of consequences calculated by Palenik/Beyea (1345 latent cancer fatalities). (Id., p. 2) The frequency of the precipitation scenario discussed by Palenik/Beyea would be about a factor of 10 lower than the frequency probability estimate calculated by the Licensees because of the low frequency of rain. (Id., pp. 1-2)

1.3-14 The effect of the combined assumptions of high health effects conversion factor, long exposure periods and relatively ineffective shielding discussed above makes the probability of the higher estimates calcu-

lated by Palenik/Beyea far lower than the 5.5×10^{-6} per reactor year calculated by the Licensees as the frequency for the lower Beyea/Palenik estimates. (Potter 1.3, p. 3) It should be emphasized that the highest estimate for latent cancer fatalities calculated by Palenik/Beyea (60,000) is substantially higher than the maximum number of these health effects calculated by the Licensees (30,000). The frequency associated with the occurrence of 30,000 latent cancer fatalities is 10^{-7} per year or less. (Potter, Tr. 8348)

1.3-15 The Licensees also estimated the probability of the large number of thyroid nodule cases calculated by Palenik/Beyea. For this health effect, the risk dominant release for the Indian Point plants is release category 2. Because the iodine release fraction for release category 2RW is lower than that for release category 2 by a factor of 7, 2RW releases do not lead to iodine radiological deposition in New York City as postulated by Palenik/Beyea. Using the same methodology described in Finding 1.3-13, the Licensees calculated an overall frequency of achieving the radiological conditions calculated by Palenik/Beyea for thyroid nodule cases of 5×10^{-8} per reactor year, or once in every 20,000,000 years, for Units 2 and 3 combined. Because the Palenik/

Beyea estimates used dose to health effect conversion factors that are too high, the probability of the lowest number of thyroid nodules calculated by Palenik/Beyea would be slightly lower than 5×10^{-8} per reactor year and the likelihood of the highest consequence estimates would be much lower than that figure. (Potter 1.3, pp. 3-4)

1.3-16 Throughout the Licensees' analysis, the source term assumptions employed in IPPSS were again used. The Board earlier found these source terms to be unduly pessimistic, in that they overstate risks. (See Finding 1.1-409) Reductions in the magnitude of the source terms, as discussed in the testimony of Licensee witnesses Stratton, Rodger and Potter, (see, Findings 1.1-276 to 1.1-307) are potentially important because they can result in large reductions in the number of latent cancer fatalities and thyroid nodules. (Potter 1.3, pp. 4-5)

A. ULTIMATE FINDING

1.3-17 The Board finds that probabilities associated with the Palenik/Beyea consequences are extremely low. Even for the lower end of the range of latent cancer fatalities calculated by Palenik/Beyea, the

probability is in the range of once in 100,000 to once in 200,000 years. The probability of ever reaching the maximum number of latent fatalities calculated by Palenik/Beyea would be much lower than that, if possible at all, because of the extremely conservative assumptions employed in their analysis. The probability of occurrence of the lowest number of thyroid nodules calculated by Palenik/Beyea is once in 20,000,000 years. Anticipated reductions in the magnitude of radiological source terms would lower even further the probability of occurrence of the radiological conditions necessary to arrive at the Palenik/Beyea consequence estimates.

V. BOARD QUESTION 1.4

1.4-1 Board Question 1.4 reads as follows:

> What risk to public health and safety is presented by the Indian Point plants through a chain of events including pressurized thermal shock (PTS) to the reactor pressure vessels?

1.4-2 Licensees and NRC Staff both presented testimony with regard to this Board question. None of the intervenors or interested states presented direct testimony.

1.4-3 Pressurized thermal shock (PTS) refers to an accidental rapid cooldown of the water in a reactor vessel while the system pressure is maintained at a high level. A rapid cooldown of the inner surface of a reactor vessel causes thermal stresses in the vessel wall. These stresses combined with pressure stresses may cause preexisting small cracks to grow larger. Reactor vessel integrity could be lost by crack extension through the vessel wall. The likelihood of crack growth under PTS conditions increases with radiation exposure, a phenomenon that is understood and monitored. Neutron irradiation of the reactor vessel wall causes the vessel materials to lose some of their initial toughness or resistance to cracking. (Woods/Klecker Testimony, p. 2; Richardson/Bley 1.4, pp. 1-2)

1.4-4 The NRC Staff is establishing a new rule which sets forth screening criteria based on a material property which is related to the fracture resistance of the reactor vessel. This material property is called the Reference Temperature for the Nil Ductility Transition (RT_{NDT}). (Woods, Klecker, Tr. 8701)

1.4-5 RT_{NDT} is a reference temperature chosen to represent the temperature at which the material

toughness (resistance to fracture) begins to increase rapidly with increases in temperature. At temperatures below the RT_{NDT}, the material is significantly less tough than at temperatures above the criterion. (Woods/Klecker Testimony, p. 3) RT_{NDT}, therefore, measures where the transition occurs between ductile and brittle behavior of metals. This transition takes place over a span of roughly 100°F. (Klecker, Tr. 8708)

1.4-6 Tests to determine RT_{NDT} are done on samples from the reactor vessel weld itself. If actual reactor vessel weld samples are not available, a special weld block is made with the identical procedures as the vessel weld, and samples are taken from that. (Woods/Klecker Testimony, p. 4; Klecker, Tr. 8711) There are documented ASME procedures to establish RT_{NDT} from the test results. (Klecker, Tr. 8714)

1.4-7 PTS risk is acceptable for those plants which have an RT_{NDT} below the screening criteria. The Staff has selected values of 270°F for axial weld materials in the reactor vessel and 300°F for the circumferential weld material. (Woods, Tr. 8701; Woods/Klecker Testimony, p. 6)

1.4-8 To establish the screening criteria, the Staff examined 350 reactor years of domestic operating experience and chose the most severe PTS events for examination. That study of operating experience was supplemented by probabilistic risk assessment analysis, including deterministic and probabilistic fracture mechanics vessel analysis. The results of the Staff analysis showed that plants whose RT_{NDT} are at the screening criteria will have a frequency of crack penetration through the reactor vessel wall of between 10^{-5} and 10^{-6} per reactor year. (Woods/Klecker Testimony, pp. 6-8; Woods, Tr. 8702)

1.4-9 The Staff analysis which determined the probability of crack penetration through the vessel wall is applicable to the Indian Point Units because: (1) five of the eight severe events studied in detail by the Staff occurred at Westinghouse plants, and (2) the NRC's probabilistic risk assessment analysis for hypothetical events is taken largely from extensive analyses of Westinghouse plants performed by Westinghouse for the Westinghouse Owners Croup. (Woods/Klecker Testimony, pp. 8-9; Woods, Tr. 8702)

1.4-10 The Westinghouse Owners Group probabilistic study upon which the NRC Staff based a portion of its analysis was a complete probabilistic risk assessment on all initiating events which might lead to low temperatures in the reactor vessel, regardless of pressure. The study was performed on 15 initiating events categories, which covered basically all of those that are presented in the IPPSS. The probabilities of all initiating events were accounted for and combined with generic fracture mechanics, looking into the actual types of materials and actual vessels. (Richardson, Tr. 8390-8391) The Westinghouse Owners Group analysis resulted in a calculated probability of 10^{-6} for crack extension without arrest and a probability of 10^{-7} for those incidents of crack extension without arrest which exceed the capability of the emergency core cooling system. (Id.)

1.4-11 Each operating nuclear plant will be required by the Staff to evaluate its specific RT_{NDT} for comparison to the screening criteria, and when they are projected to be within three years of exceeding the criteria, they will be required to do a plant-specific safety analysis to assess PTS risk. This will assist in identifying necessary corrective actions. (Woods, Tr. 8701-8702)

1.4-12 The Licensees both submitted analyses comparing their plant design with the generic Westinghouse probabilistic risk assessment of PTS. These analyses examined whether any plant-specific features at Indian Point would make PTS more or less likely than that determined in the generic study. (Woods, Tr. 8724) As of December 1981, Indian Point Unit 2 was at an RT_{NDT} value of 189°F, compared to the screening criterion for the axially-oriented weld of 270°F. (Woods/Klecker Testimony p. 9) The screening criterion for the axially-oriented welds at Indian Point is the governing criterion for these plants. (Id., p. 6) Based on the December 1981 RT_{NDT} value of 189°F at Indian Point 2, the facility is in the range of ten years behind the plants of greatest concern with regard to the day at which they will exceed the NRC Staff screening criterion for sensitivity to PTS. (Woods/Klecker Testimony, p. 5; Woods, Tr. 8704)

1.4-13 PTS risk decreases by approximately an order of magnitude for each 40°F reduction in RT_{NDT}. On that basis, as of December 1981, the PTS risk at Indian Point is 1 to 2 orders of magnitude below the previously stated 10^{-5} to 10^{-6} frequency range for crack extension without arrest for a plant at the screening crite-

rion. Therefore, the probability of crack extension without arrest at Indian Point is, at the present time, in the range of 10^{-6} to 10^{-8}. (Woods/Klecker Testimony, p. 9; Woods, Tr. 8740)

1.4-14 The IPPSS evaluated the frequency of a reactor vessel rupture large enough to exceed the capability of the ECCS. This evaluation yielded a mean frequency of 3×10^{-7} per reactor year for all types of vessel failure, which included those failures identified as PTS events. The quantitative results of the NRC Staff and Westinghouse Owners Group analyses are bounded by the IPPSS evaluation. (Bley/Richardson 1.4, p. 2)

1.4-15 The analysis of pressurized thermal shock risk arbitrarily assumes the existence of a crack in the reactor vessel wall even though none has been found and none is believed to exist. (Woods, Tr. 8718) This assumption of the existence of a crack is a conservatism in the analysis. (Woods, Tr. 8740) Because some crack sizes, shapes and locations will not preclude the ability of the emergency systems to keep the core cooled, not all through wall cracks will result in core melt. Therefore, the core melt frequency due to PTS events is lower than the stated frequency of PTS-related crack extension

without arrest. (Woods/Klecker Testimony, p. 8) Further, even if crack extension through the vessel wall were to cause a vessel rupture, this could cause a core melt but would not by itself cause loss of containment integrity. Additional unlikely events, such as loss of containment heat removal, would need to simultaneously occur before a significant release of radioactivity would be possible. (Richardson/Bley 1.4, p. 3; Bley, Tr. 8383)

1.4-16 Ultrasonic testing is used to detect cracks, and inspection occurs at approximately ten-year intervals. Techniques currently in use can detect flaws as small as one-half inch in depth. (Klecker, Tr. 8718-8719) The critical cracks, i.e., the cracks which are likely to grow under PTS, are one-half to one and one-quarter inches in depth. (Klecker, Tr. 8719)

1.4-17 When the Indian Point plants are operating at one hundred percent power, the RT_{NDT} values increase approximately 7°F per year. In order to assure that the screening criterion will not be reached before the end of life of Indian Point Unit 2 and that the PTS risk remains acceptable through the end of life, the Staff has encouraged many PWRs, including Indian Point, to take actions that are reasonably practicable to reduce flux

(fast neutron flux) at the pressure vessel wall to slow the rate of increase of RT_{NDT}. For Indian Point 2, a flux reduction by a factor of approximately 1.4 would assure that the screening criterion would not be exceeded until beyond the end of life of Indian Point 2. (Woods/Klecker Testimony, pp. 9-10; Woods, Tr. 8703)

1.4-18 One method of achieving such a flux reduction is the implementation of a low-leakage fuel loading pattern (LLLP). (Woods, Tr. 8742) A standard fuel loading pattern consists of placing only new fuel assemblies at the core periphery during each refueling. The LLLP program instead utilizes a fuel loading pattern which strategically places some irradiated (and therefore less reactive) fuel assemblies at the core periphery. This reduces the number of neutrons emitted from the core periphery, achieving lower neutron irradiation of the reactor vessel wall. (Lee Testimony, pp. 1-2; Woods, Tr. 8742)

1.4-19 After studying several possible low-leakage fuel loading arrangements, Con Edison directed Westinghouse to incorporate an LLLP into the core design for Fuel Cycle 6, which commenced operation in January 1983. (Lee Testimony, p. 2) Con Edison presently intends

to continue LLLP throughout the remaining service life of the plant. (Lee Testimony, p. 2) Based upon the Indian Point 2 specific analysis performed by Westinghouse, the LLLP design incorporated in Cycle 6 will reduce fast neutron flux at the point of peak vessel wall exposure by 44% compared to a standard fuel loading pattern. (Lee Testimony, p. 2)

1.4-20 The Indian Point 2 LLLP design is projected to give approximately the same peak vessel wall exposure at the end of 32 Effective Full Power Years (EFPYs) of operation as the standard loading pattern at the end of 20 EFPYs. Actual neutron flux measurements made at the beginning of Cycle 6 operation with the neutron detectors located outside the vessel wall were found consistent with the above projections. (Lee Testimony, pp. 2-3) The RT_{NDT} values at the end of life of Unit 2, utilizing the LLLP design, will be well below the NRC's screening criteria for the circumferential and axial flaws. (Lee Testimony, p. 3)

A. ULTIMATE FINDING

1.4-21 Based on the testimony and cross-examination regarding this Board Question, the Board finds that

there is reasonable assurance that the risk to public health and safety from an accident initiated by PTS at the Indian Point units, is substantially less than those events which dominate risk, and will remain so for the balance of the service lives of the units.

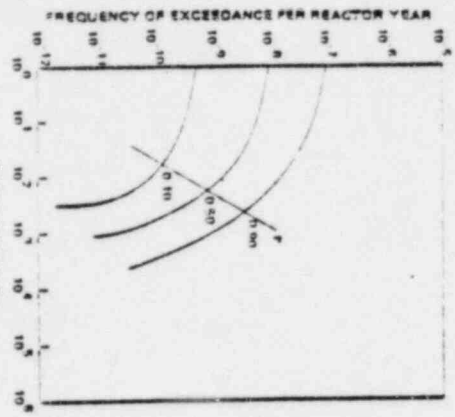


Figure III-1a

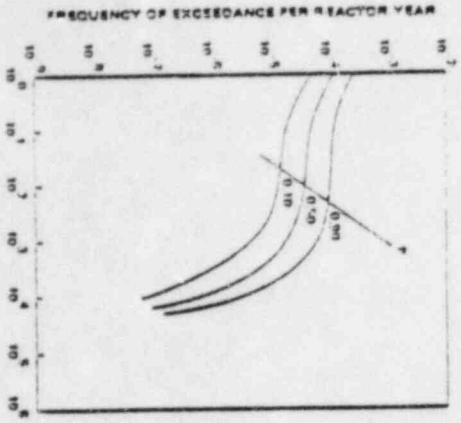


Figure III-1d

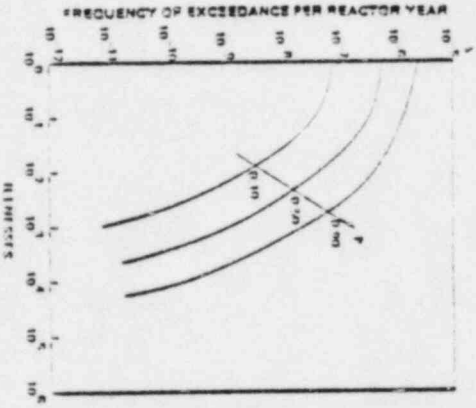
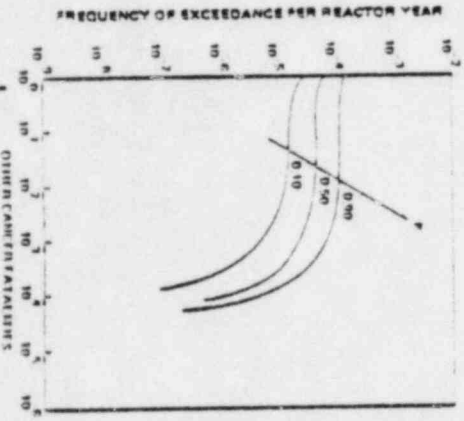


Figure III-1b



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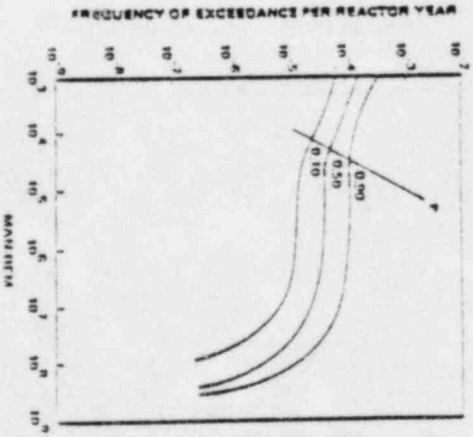

Figure III-1c

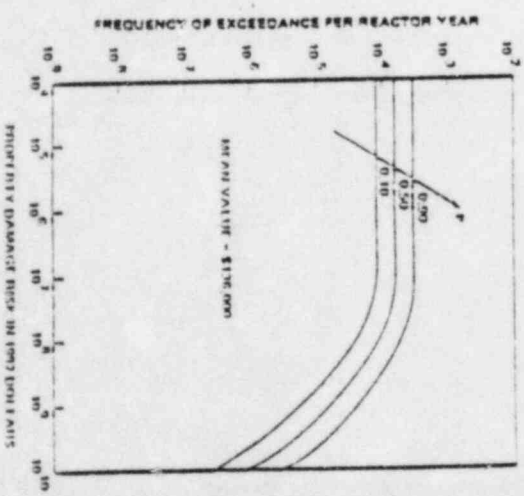

Figure III-1f

Figure III-1. Indian Point 2
Societal Risk Curve

APPENDIX B

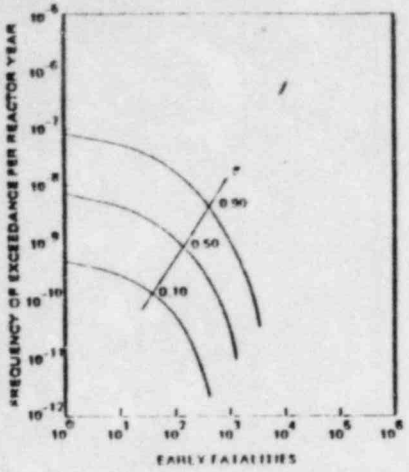

Figure III-2a

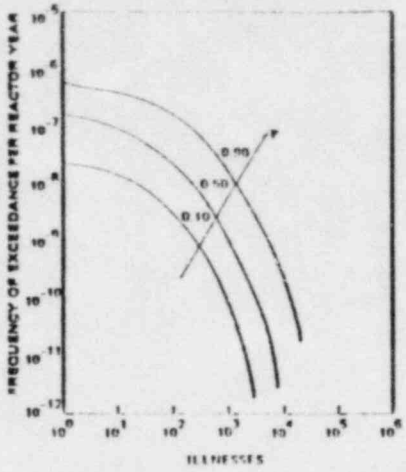

Figure III-2b

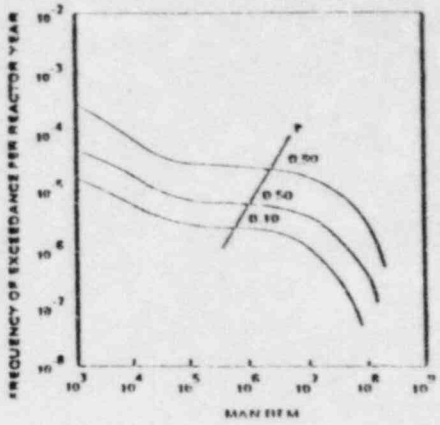

Figure III-2c

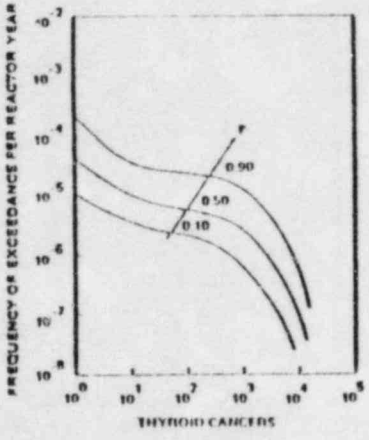

Figure III-2d

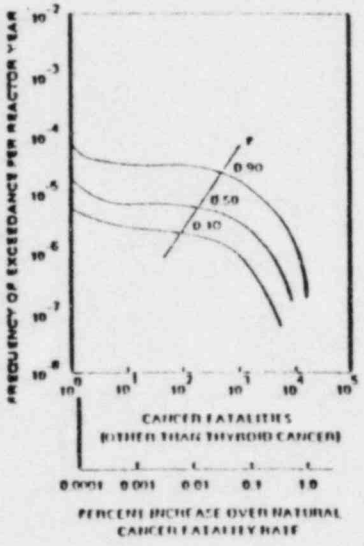

Figure III-2e

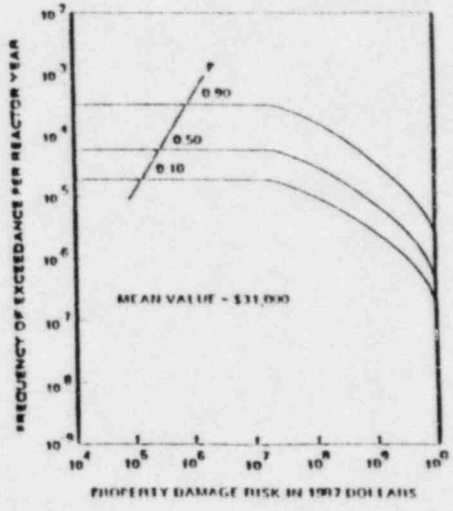

Figure III-2f

Figure III-2. Indian Point 3
Societal Risk Curve

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

Docketed USNRC
JUL 13 1983
Office of the Sec.
Docketing & Sec.
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) : July 11, 1983

CON EDISON'S PROPOSED FINDINGS OF FACT REGARDING COMMISSION QUESTION 2

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COMMISSION QUESTION 2

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I. INTRODUCTION

2.0-1 Commission Question 2 reads as follows:

> What improvements in the level of safety will result from measures required or referenced in the Director's Order to the licensee, dated February 11, 1980? (A contention by a party that one or more specific safety measures, in addition to those identified or referenced by the Director, should be required as a condition of operation would be within the scope of this inquiry if, according to the Licensing Board, admission of the contention seems likely to be important to resolving whether: (a) there exists a significant risk to public health and safety, notwithstanding the Director's measures, and (b) the additional proposed measures would result in a significant reduction in that risk.)

2.0-2 Pursuant to the Commission's directions regarding the two-pronged test set forth in the question, the Board ultimately admitted three contentions as issues in controversy. In addition, the parties were invited to submit testimony concerning one Board question. [Memorandum and Order (Formulating Final Contentions and Setting Schedule), dated November 15, 1982] The findings of fact below will treat the Commission's question first and then the Board question and contentions seriatim.

II. THE DIRECTOR'S ORDER

2.0-3 The Director's Confirmatory Order (the Order) of February 11, 1980 was issued with the intent of

providing additional assurance of safe operation of the Indian Point plants while additional studies were underway to investigate their safety. (Rowsome/Hannon 2, p. 2) However, at the time of the Order, no plant-specific PRA of the Indian Point plants had yet been performed. Therefore, in imposing the requirements contained in the Order, the Director concentrated on principal contributors to risk identified in previous generic risk studies, as well as lessons learned from previous reactor accidents such as the one at Three Mile Island. Since these previous risk studies had not analyzed risk contribution from external initiating events, the measures contained in the Director's Order primarily focused on reducing the risk from internal initiating events. (Bley/Richardson 2, p. 2)

2.0-4 The requirements of the Order fall into five basic areas. First, in the area of conduct of operations, surveillance testing and maintenance, the Order was intended to make transients less frequent, selected safety systems more reliable, and emergency response more comprehensive and reliable. Second, in the area of staffing, the Order required that two rather than one Senior Reactor Operator be in the control room of each plant, on each shift, and that the pool of safety consultants available to each Licensee be expanded. Third, the Order required

that operators of the facilities and on-site emergency response teams undergo expanded and accelerated training for severe reactor accidents as well as for normal operation. Fourth, engineering margins related to limiting core temperature excursions during large LOCA accidents were increased. Finally, additional studies of the susceptibility of the plants to severe accidents were ordered to increase Licensee understanding of the susceptibility of the facilities to severe accidents and to explore options for further risk reduction. (Rowsome/Hannon 2, p. 3) The Board finds, based upon the uncontroverted testimony of NRC Staff witnesses that the Licensees have complied or were, at the time of the hearings, in the process of complying with each requirement of the Order. (See Tr. 6582- 6663)

2.0-5 Although Licensees' compliance with the provisions of the Order has reduced risk, no quantitative expression can be made of the extent to which risk has been reduced by the Order. There are several reasons for this. First, although the Order relied on previous generic risk studies which had not focused on external initiating events, the IPPSS performed by consultants to the Licensees indicated that external initiating events were the major contributors to risk at the Indian Point facili-

ties. (Bley/Richardson 2, p. 3; Rowsome/Hannon 2, p. 5) Therefore, none of the items in the Director's Order had specifically addressed the dominant scenarios with respect to risk which had been identified for the Indian Point plants in the IPPSS. (Bley, Tr. 6443) Second, the Staff testified that it was unable, using present PRA techniques, to devise a reliable quantitative measure for the change in risk resulting from the adoption of the staffing and staff training measures contained in the Order. (Rowsome/Hannon 2, pp. 4-5; Rowsome, Tr. 6573, 6665) Third, the improvements made at the Indian Point facilities as a result of the Order represent but a portion of the overall design and operational improvements made at the facilities during and after the imposition of the requirements of the Order. In particular, as a result of the analysis contained in the IPPSS, Licensees have made extensive modifications in equipment and procedures independent of the requirements of the Order. There is considerable difficulty, therefore, in separating out the impact on the level of safety attributable to the Order from that attributable to the balance of the modifications made at the facilities as a result of the IPPSS analysis. Efforts to distinguish quantitatively between those improvements mandated by the Order and those independently undertaken by the Licensees were generally not attempted by either

the Licensees or the Staff. (Rowsome/Hannon 2, p. 4; Bley/Richardson 2, pp. 3-5)

2.0-6 The only modifications to the facilities mandated by the Order for which risk reduction was quantified were those improvements related to safety system design, surveillance testing and changes in technical specifications. The Staff estimated that these changes would result in a factor of three reduction in the frequency of accident sequences involving the interfacing systems LOCA, loss of offsite power and reliability of the auxiliary feedwater system. (Rowsome/Hannon 2, p. 5; Rowsome, Tr. 6577, 6665, 6669) The overall risk reduction attributable to changes in safety system design at the facilities was estimated by the Staff to be less than a factor of three based on the identification in the IPPSS of accident sequences triggered by external events for which these changes make little or no difference. (Rowsome/Hannon 2, p. 5; Rowsome, Tr. 6573, 6665) No other party took issue in their testimony with the general conclusions of the NRC Staff or the Licensees regarding the impact of the Director's Order requirements.

2.0-7 Although the reduction in plant risk attributable to the requirements of the Order cannot be

precisely quantified, improvements resulting from the Order are applicable to most if not all of the internally initiated accident sequences to which the plants may be subject. (Rowsome, Tr. 6665) As a result of these modifications and procedures, the likelihood that operators might fail to diagnose a severe reactor accident promptly and correctly is smaller today than it was three or four years ago. (Rowsome/Hannon 2, p. 5)

A. ULTIMATE FINDING

2.0-8 Based on the direct testimony and the cross-examination thereon, the Board finds that the overall improvements in the level of safety attributable to the measures required by the Order of February 11, 1980 can best be characterized, as the Licensees have characterized it, as modest but positive. (Bley/Richardson 2, p. 3)

III. NECESSITY OF ADDITIONAL SAFETY MEASURES

2.0-9 The Board has already found that the risks associated with the operation of the Indian Point units are very low and meet the Commission's public health risk safety goals. In addition, the risks posed by operation of the Indian Point Facilities are within the range of risks posed by other operating nuclear plants. (See

Findings 1.1-407, 1.1-408, 1.1-409, 5.0-76) Therefore, we find that no additional safety devices are warranted. Accordingly, discussion of specific additional safety features proposed by the intervenors in this proceeding is unnecessary. However, since the Board has been directed to gather and present to the Commission all relevant information, the Board finds it appropriate to include a discussion of the safety measures proposed.

IV. CONTENTIONS 2.1(a) and (d)

2.1-1 Contention 2.1(a) reads as follows:

> A filtered vented containment system for each unit must be installed.

2.1-2 Contention 2.1(d) read as follows:

> A separate containment structure must be provided into which excess pressure from accidents and transients can be relieved without necessitating releases to the environment thereby reducing the risk of containment failure by overpressurization.

A. PROFESSIONAL QUALIFICATIONS OF UCS/NYPIRG WITNESSES

2.1-3 Both Mr. Sholly and Dr. Thompson, the witnesses for UCS/NYPIRG with regard to these contentions, were questioned concerning their professional qualifications. (Tr. 6148-6165)

2.1-4 Mr. Sholly has a Bachelor of Science degree in Education with a double major in Earth and Space Science and General Science. He also has a minor in environmental education. He has taken graduate courses in Land Use Planning. (Tr. 6148)

2.1-5 Mr. Sholly has no engineering training, no training in the engineering design of filtered vented containment systems and no other experience with them. (Tr. 6149, 6153, 6155) Mr. Sholly's knowledge of filtered vented containment systems is limited to his review of the literature with respect to that subject and his discussions with Dr. Thompson. (Tr. 6155)

2.1-6 Mr. Sholly testified that he has experience in the general area of environmental impact, (Tr. 6155), and that the particular relevance of his training to this testimony revolves around the environmental impact of radiation. (Tr. 6156) Mr. Sholly's training in the impact of radiation is limited to a small portion of a fifteen week, three credit course on Principles of Ecology which he took in his sophomore year of undergraduate school. (Tr. 6157) He has had no other formal training in environmental impacts of radiation. (Id.)

2.1-7 The Licensees and the NRC Staff challenged the qualifications of Mr. Sholly to testify with regard to filtered vented containment systems (FVCS) and separate containment systems (SCS), but following a liberal interpretation of the Federal Rules of Evidence, Rule 702, the Board determined that Mr. Sholly should be permitted to testify regarding these contentions. (Tr. 6163)

2.1-8 Dr. Thompson testified that he has a mechanical engineering background, but that this did not place as much emphasis on the properties of structures as would have been the case in a civil engineering program. (Tr. 6164) Dr. Thompson further testified that he had no direct familiarity with the IPPSS and had made no independent analysis of the Indian Point design. (Tr. 6206)

2.1-9 The Board finds that Mr. Sholly's lack of formal training in engineering and the environmental effects of radiation requires us to accord his testimony less weight than that of other witnesses. Likewise, although Dr. Thompson has had extensive training in physics, his limited knowledge of civil or structural engineering and lack of familiarity with the IPPSS and the Indian Point plant-specific design reduce the weight of his testimony.

B. IDENTIFICATION OF PROPOSED ADDITIONAL SAFETY FEATURES

2.1-10 An FVCS is a device incorporated into the design of a containment building which allows for controlled venting of the containment atmosphere. Controlled filtered venting is a process in which a portion of the containment atmosphere would be deliberately released to the environment in a controlled manner through a system of filters and energy absorbers. Such a pressure relief system would be actuated to reduce containment pressure, a pressure which could under some severe accident conditions otherwise lead to containment failure and thereby allow the uncontrolled and unfiltered release of radionuclides into the atmosphere. This system would only be used to mitigate the effects of an accident which is beyond the design basis accident. (J. Meyer 2.1, p. 6; Sholly/Thompson Testimony, p. 11)

2.1-11 An SCS would provide more containment volume and attendant heat sink and source-term attenuation capability by means of interconnecting piping from the existing containment building to another large building. As with an FVCS, this concept may have the potential to

prevent or delay containment building failure and thereby reduce consequences to the public from severe overpressurization accidents that would have otherwise resulted in containment failure. (J. Meyer 2.1, p. 14; Sholly/Thompson Testimony, pp. 11-12)

C. PREVENTION vs. MITICATION OF ACCIDENTS

2.1-12 UCS/NYPIRG witnesses Sholly and Thompson testified that core melt accidents dominate risk at nuclear plants in general and Indian Point in particular. (Sholly/Thompson Testimony, pp. 3-4) The witnesses further testified that while attempts have been made to reduce the probability of core melt events, these attempts have failed to afford sufficient risk reduction to render mitigative features unnecessary. (Id., p. 6) As support for this proposition, the witnesses cited the Director's Order, the effect of which was to reduce the probability of core melts by a factor of three or less. (Id., pp. 6-7; see Finding 2.0-6)

2.1-13 These witnesses concluded that the prospects for further reduction in risk by addressing probability are limited, and that because they believe core melt events still have a relatively high probability, reduction

of accident consequences appears to offer a greater opportunity for risk reduction than does reduction of accident probabilities. (Sholly/Thompson Testimony, pp. 4-5)

2.1-14 While UCS/NYPIRG is correct that core melt accidents dominate risk at nuclear facilities, this is because only core-melt events provide a theoretical mechanism for releasing a large fraction of the radioactive inventory from the core to the environment. (Bley/Richardson 2.1, p. 3) Thus, no matter how much the probability, or for that matter, the consequences of such events are reduced (even with the implementation of a filtered vented or separate containment system), the effects of core melts can never be reduced to the low level of non-core melt events. In that sense, core melt events will always dominate risk. (Bley, Tr. 6300, 6302) However, as UCS/NYPIRG conceded, there is no risk to the public represented by core melt events which do not result in containment failure or bypass. (Thompson, Tr. 6193) Indeed, at Indian Point, core melt frequency is a very poor indicator of risk. (Bley/Richardson 2.1, p. 8; Richardson, Tr. 6394; see Finding 1.1-127) For example, the early fatality risk comes almost entirely from one sequence, the interfacing systems LOCA. (Bley/Richardson 2.1, p. 8; Bley, Tr. 6377, 6401) This accident bypasses

containment and thus would bypass an FVCS or SCS. (See Finding 1.1-369) Yet, this accident contributes less than 0.5% to the overall core melt frequency for Indian Point. (Bley/Richardson 2.1, p. 8) Thus, there is no correlation between magnitude of core melt frequency and magnitude of risk. (Richardson, Tr. 6299) The NRC Staff also testified that the frequency of severe releases is a better measure of offsite radiological risk than is the overall core melt frequency. (Rowsome, Tr. 12912)

2.1-15 Contrary to UCS/NYPIRG's testimony, beyond the benefits in risk reduction realized by compliance with the Director's Order, reductions in risk achieved by addressing probability of core melt have already occurred at Indian Point. (Bley/Richardson 2.1, pp. 3-4; Rowsome/Blond 5, p. C-30) During the course of performing the IPPSS analysis, certain areas were identified where improvements in equipment or procedures were appropriate. (Bley/Richardson 2, p. 4; Bley/Richardson 2.1, p. 4; Bley, Tr. 6282) Appropriate modifications were then undertaken at both Indian Point units including:

1. Implementation of a refueling interval surveillance test to verify disc integrity for RHR valves MOV-730 and MOV-731.

2. Implementation of a refueling interval surveillance test to verify disc to stem integrity for spray valve 869A (such verification was already being accomplished for redundant valve 869B in the course of performing normal refueling operations).

3. Implementation of changes to refueling interval surveillance test to clarify test method and data recording for accumulator discharge valve flow verification check.

4. Implementation of procedures to periodically verify that: (a) the service water pumps designated for essential header service are in fact aligned to the required essential services (i.e., field valve alignment check), and (b) the position of the control room service water system mode selector switch is such that the correct set of service water pumps is selected for automatic essential header service.

5. System modification, procedural change or verification testing to ensure that sufficient backpressure will be maintained in the service water system to prevent service water pump overload for cases in which only one service water pump is operating with the system in accident configuration.

(Bley/Richardson 2, p. 4)

2.1-16 Con Edison has made the following additional improvements at Indian Point 2:

1. Rearrangement of diesel generator fuel oil transfer pump power supplies such that the primary transfer pump for each diesel is powered from one of that diesel's electrical buses.

2. Replacement of manual isolation valves with motor operated isolation valves in certain of the fan cooler service water discharge lines.

3. Seismic cushioning of the control building roof.

4. Enhancement of installed alternate shutdown system.

(Bley/Richardson 2, p. 5)

2.1-17 These changes were voluntarily implemented by the Licensees and yield a reduction in the core melt frequency and risk of operating these facilities.

(Bley/Richardson 2.1, p. 4; see Rowsome/Blond IV.A, Figures IV.A.1 and IV.A.2; Budnitz Testimony, Table 2, p. 36; Buchbinder/Kubicki Testimony, Table 4, p. 23) Even if further plant modifications were deemed necessary, prevention fixes might still be attractive alternatives to mitigation. (Rowsome/Blond 5, p. C-15)

2.1-18 The Board finds that, contrary to the position taken by UCS/NYPIRG, risk reduction can be achieved not only by the implementation of accident mitigating measures but also, as the Licensees have demonstrated, by adopting measures which result in meaningful reductions in the frequency of severe accidents.

D. CONTAINMENT STRENGTH

2.1-19 The testimony and cross-examination regarding Contentions 2.1(a) and (d) focused to a significant degree on the uncertainties regarding the strength of the Indian Point containments. The pressure capability of the Indian Point containments is obviously a crucial issue, since, as UCS/NYPIRG has conceded, the higher the containment failure pressure, the less the benefit which can be expected from an FVCS or SCS. Obviously, the higher the containment failure pressure, the fewer the

number of accident sequences which would fail the containment; the usefulness of an FVCS or SCS would, therefore, be reduced. (Thompson, Tr. 6207)

2.1-20 Based on the evidence and cross-examination, the Board has found that no breach of the Indian Point containments would occur at pressures below 141 psia. (See Finding 1.1-408) Only one of the ten containments reviewed by the Licensees' consultant has a pressure retaining capability greater than that of the Indian Point containments. (See Finding 1.1-147) This pressure retaining capability substantially reduces the potential usefulness of the mitigative devices proposed by UCS/NYPIRG for the Indian Point plants.

E. ENGINEERING FEASIBILITY OF FVCS and SCS

2.1-21 UCS/NYPIRG proposed no detailed design features for an FVCS or SCS at Indian Point. Witnesses Thompson and Sholly did suggest a gravel filter concept similar to the one intended for installation at the Barseback Nuclear Plant in Sweden. The witnesses recommended that the operator be provided with control over the

valve between the containment and the filtered venting system to provide flexibility in determining when to initiate venting based on existing accident conditions. The witnesses also suggested that an automatic initiation feature be included in the design to assure that filtered venting is initiated once the containment pressure reaches a preset value. (Sholly/Thompson Testimony, p. 12)

2.1-22 As an alternative to an FVCS, UCS/NYPIRG suggested an SCS which would incorporate a pressure suppression pool and hydrogen igniters which would provide for hydrogen control in the secondary containment in an attempt to protect against compartment failure as a result of hydrogen burns. Provision for both manual and automatic initiation of the SCS was proposed. (Sholly/Thompson Testimony, pp. 12-13)

2.1-23 UCS/NYPIRG maintained that a separate FVCS or SCS must be employed for each unit at the Indian Point site due to the risk of accidents at both plants arising from a common event. (Sholly/Thompson Testimony, p. 13)

2.1-24 Technology concerning filtered venting and separate containment systems is immature and unprov-

en. There has been no detailed containment phenomena study, probabilistic risk assessment, or sound design evaluation and testing program to verify that these mitigation systems would operate successfully under the specific conditions of interest. (Bley/Richardson 2.1, p. 19; Bley, Tr. 6381-6382) Licensee and Staff witnesses acknowledged that while there are hundreds of reactor years of operating experience and experience from pressure and leak tests with large pressurized water reactor containments, there is no such experience with either filtered vented or separate containment systems installed on operating reactors. (Bley/Richardson 2.1, p. 9; Meyer, Tr. 6841)

2.1-25 No FVCS or SCS exists in any commercial nuclear power plant in the United States. (Bley/Richardson 2.1, p. 9; J. Meyer, Tr. 6841) Two operating test reactors, the Zero-Power Plutonium Reactor (ZPPR) test facility and the Fast Flux Test Facility (FFTF), have installed or are installing an FVCS, and portions of the research and development would be applicable to large, dry PWRs such as Indian Point. (J. Meyer 2.1, p. 9; J. Meyer, Tr. 6841, 6842) The German SNR-300 prototype liquid metal fast breeder reactor has an FVCS or is planning to install one. (J. Meyer 2.1, pp. 9-10) An SCS exists in several heavy water nuclear power plants in Canada. (Id., p. 15;

Thompson, Tr. 6201) The French are considering installing an FVCS in their existing 900 Mw electric PWRs. (J. Meyer 2.1, p. 11) However, the relevance to Indian Point of the proposed devices at these facilities has not been demonstrated. (See J. Meyer, Tr. 6841-6842, 6874) In fact, the French have done little work in analyzing degraded core accidents, (Richardson, Tr. 6380), and the type of SCS installed in the Canadian reactors is not the type UCS/NYPIRG recommends for Indian Point. (Thompson, Tr. 6201-6202)

2.1-26 Although an FVCS has been proposed for the Clinch River Breeder Reactor, a liquid metal cooled reactor, neither UCS/NYPIRG nor the NRC Staff demonstrated its relevancy to Indian Point. (See Sholly/Thompson Testimony, p. 16; J. Meyer 2.1, p. 10) The technology regarding the accommodation of fission products at Clinch River is not relevant to PWR designs. Moreover, it may or may not be more difficult to design and construct an FVCS for a fast breeder reactor than for a PWR. (J. Meyer, Tr. 6874) In addition, the materials requiring filtration in the containments of sodium-cooled reactors such as Clinch River and the FFTF are in some instances dramatically different. (J. Meyer, Tr. 6842)

2.1-27 UCS/NYPIRG's assertion that FVCS and the SCS are feasible from an engineering standpoint is in large part based upon the fact that an FVCS has been planned and designed for the Barseback reactors in Sweden. (Sholly/Thompson Testimony, pp. 15, 16) In light of the testimony and cross-examination, the fact that filtered vented containment systems are planned for that reactor appears to have questionable applicability to their possible implementation in the near term at Indian Point. The Barseback FVCS is not operational. Further, the Barseback reactors are boiling water reactors having a containment volume less than one half that of the Indian Point facilities. (Thompson, Tr. 6199) Volume is an important consideration in determining the need for an FVCS in that all else being equal, the smaller the containment volume, the earlier the containment will fail. Indeed, the ratio of the amount of steam released to the containment size could make failure of a boiling water reactor containment more likely than that of a pressurized water reactor containment such as at Indian Point. (Thompson, Tr. 6199, 6200) The Licensees testified that the decision to employ an FVCS at the Barseback reactors was not based upon a detailed background analysis to determine whether such a system would really assist in degraded core situations or whether there was undue risk to the

public which justified its implementation. (Richardson, Tr. 6380, 6382, 6391) Because of the fact that Barseback is a boiling water reactor with a different volume and pressure and different composition of gases vented than the Indian Point facilities, it cannot be said with assurance that the Barseback FVCS design work would be applicable to Indian Point. (Thompson, Tr. 6224)

2.1-28 Given the differences between the types of facilities discussed in the UCS/NYPIRG and NRC Staff testimony and the Indian Point facilities, Staff witness Meyer agreed with the Licensees that although much generic research and development work has been done on filtered vented containment systems, for Indian Point one would have to prepare a detailed plant-specific design, which the Staff had not done. (Tr. 6846) Dr. Meyer further testified that much of the technology with regard to filtered vented containment systems which has been done previously would not be applicable to light water pressurized water reactors with large dry containments, such as Indian Point. (Tr. 6874)

2.1-29 All of the parties agreed that there are potential problems from a regulatory standpoint. Specifically, there is no regulatory guidance on the design,

licensing, operation or testing of either an FVCS or SCS. (Bley/Richardson 2.1, p. 19; Sholly, Tr. 6221-6222) Further, in implementing any FVCS design, one would be faced with a design which involves an intentional breach in the containment boundary, and this is a departure from past practice at the NRC. (Meyer, Tr. 6845)

2.1-30 The Licensees presented a preliminary check list of design requirements, which would have to be taken into account in the design and review of a filtered vented containment system. This list was taken from NUREG/CR-1410, a document on which UCS/NYPIRG relies. These design considerations include functional requirements for the system, operational requirements, requirements related to resistance to hazard, requirements related to reliability and system control, sabotage protection, and considerations related to inspection, testing, maintenance, and post-accident safety and repair/restoration. (Bley/Richardson 2.1, pp. 14-17) Further, no regulatory guidance has been issued with regard to the design, licensing, operation or testing of either an FVCS or SCS. (Id., p. 19)

2.1-31 The Licensees' witnesses were cross-examined in some detail with regard to a filtered source

term presented in the IPPSS. The Licensees testified that release category 2RV of the IPPSS was simply a partial, preliminary effort to compare a vented release with an unvented release, and not a comparison of a plant with a vented system to a plant without a hypothetically assumed vented system. (Richardson, Tr. 6380) This preliminary effort did not entail a full analysis of initiating events, containment performance, or a detailed consideration of the factors listed in NUREG/CR-1410. (Bley/Richardson 2.1, p. 19; Bley, Tr. 6303) The Licensees do not claim that the source term discussed in the IPPSS as the 2RV category would be an appropriate source term for any particular design of filtered vented containment system. (Bley, Tr. 6310-6311) In addition, the IPPSS discussion of release category 2RV assumed that no failure of the filtered vented system had occurred. (Bley, Tr. 6307) The discussion of a filtered source term in the IPPSS was characterized by the Licensees' witnesses as an academic exercise by the source term experts who had prepared this section of the study. (Richardson, Tr. 6306) While UCS had the opportunity to pursue this matter with the source term experts during the consideration of Commission Question 1, counsel for UCS declined to do so.

2.1-32 A report entitled "Post-Accident Containment Vent Systems Conceptual Designs" was prepared by the Water Reactors Division of Westinghouse in March 1980. This report contained only conceptual designs of an FVCS. Licensee witness Richardson, a Westinghouse employee, testified that this report was a very brief scoping study done over a period of sixty days as a precursor to the Zion probabilistic risk assessment. The report was based upon the WASH-1400 release categories and failure pressure, (Richardson, Tr. 6341, 6383), and was prepared before any plant-specific core-melt, core/containment analysis had been performed for the Indian Point facilities. No functional requirements for a filtered vented containment system were identified in the report. (Richardson, Tr. 6338) Based upon these considerations Licensees testified that the report had no usefulness in terms of any FVCS design function. (Richardson, Tr. 6383, 6384)

2.1-33 The NRC Staff was also cross-examined with regard to the value of the Westinghouse report to an understanding of the feasibility of filtered vented containment systems. Staff witness Meyer testified that the study was a good first try at considering conceptual designs for an FVCS, (Tr. 6875), but that it had no currency

or relevance to the Staff's position in the hearing regarding whether this mitigative feature should be required for the Indian Point facilities. (Tr. 6877, 6880)

F. POTENTIAL USE OF INDIAN POINT 1 CONTAINMENT AS AN SCS

2.1-34 Since Indian Point Unit 1 is no longer in operation, the NRC Staff has examined the feasibility of using the Unit 1 containment as an SCS. (J. Meyer 2.1, p. 16) Some of the information which would be necessary to make a finding of feasibility for the use of Unit 1 as a separate containment system is lacking. In addition, feasibility is a strong function of the effect of external events (hurricanes, earthquakes, floods and tornadoes) on the Unit 1 containment. However, with these qualifiers, the Staff views the use of the Unit 1 containment as a separate containment system to be feasible to significantly delay or reduce the effects of severe overpressurization accidents caused by internal event initiators and fires. The Staff does not believe that it is feasible to use the Unit 1 containment to accommodate severe overpressurization accidents caused by seismic external event initiators because the seismic resistance of Indian Point 1 is substantially less than either of the other two units.

The Staff does not have sufficient information available to assess the capability of the Unit 1 containment with regard to other external initiating events. (J. Meyer 2.1, p. 17)

2.1-35 There are both attractive and unattractive features of the Unit 1 containment which bear upon its possible use as an SCS at Indian Point. Unit 1 is a substantial structure which has considerable pressure/passive heat sink capability, together with potential active cooling capability via containment sprays, both internal and external to the containment boundary. The large volume and surface area could increase plate-out and condensation effects and thereby reduce the radioactivity available for release to the environment. However, the facility was shut down in 1974, and the effect of eight years of disuse is unknown. In addition, some safety systems have been dismantled, and the cost of rehabilitating them is unknown; unless the active containment cooling systems described above are functional, the containment building has limited ultimate heat sink capability. Further, the potential combustion of hydrogen transported to the Unit 1 containment under continued venting could require the addition of other mitigation devices and further consideration of the structure's pressure reducing capabilities. (J. Meyer 2.1, pp. 18-20)

2.1-36 While there appear to be sufficient containment penetrations available to accommodate the necessary vent piping from Units 2 and 3 to Unit 1, the design of those lines would require substantial hardening against external events, which could be very costly. Indeed, Unit 1 was originally designed to a seismic capability considerably below that of Units 2 and 3, and it may not have been specifically designed to meet other external event design criteria which were imposed on the other two facilities. Thus, Unit 1 may not be a worthy candidate for mitigation of core melt events caused by external initiators. (J. Meyer 2.1, pp. 18-20) This is particularly significant, since external events are the dominant contributors to late overpressurization scenarios. (See Findings 1.1-379, 1.1-380, 1.1-381)

G. FAILURE MODES AND ATTENDANT RISKS

2.1-37 In considering mitigation systems, the NRC Staff testfied that a substantial loss in risk reduction capability occurs when "ideal" mitigation assumptions are replaced with those which consider unreliability and attendant risks. (Meyer/Pratt Testimony, p. III.B-40) Attendant risks are new risks which result from imple-

mentation of a mitigation feature. (J. Meyer, Tr. 6733-6734) Although many possible failure modes and attendant risks associated with FVCSs and SCSs have been identified, (see Findings 2.1-38 to 2.1-46) it should be noted that no design has been perfected to the point where the failure modes could be evaluated successfully or even evaluated as to the amount of effort required to engineer the failure modes out of the system. (Bley, Tr. 6338)

2.1-38 One of the principal disadvantages of any filtered vented containment system is that the system can fail to operate properly or, even if it does operate properly, can cause the failure of other safety features by adverse systems interaction. As a result, the installation of an FVCS would under certain circumstances release radionuclides that would not otherwise have been released, exacerbating normally contained low-consequence accidents. (J. Meyer 2.1, p. 8) Any detailed analysis of a filtered vented containment system or separate containment system would have to take into account the types of failure modes that are possible, and the timing, pressure, temperature and types of energy releases that the mitigation device would have to handle. (Richardson, Tr. 6380-6381)

2.1-39 There was little disagreement among the parties concerning the types of failure modes which an FVCS or SCS would have to be designed to accommodate. All parties agreed that the mitigation device would have to be designed to accommodate the external events which are the dominant contributors to risk at the Indian Point facility. (Bley, Tr. 6358; J. Meyer, Tr. 6695, 6785, 6786; Sholly, Tr. 6185, 6228) Unless the mitigation device was designed to withstand events (seismic, fire, wind) which dominate the risk at Indian Point, a potentially late overpressurization failure of the containment could become an early release scenario which would not only add property damage and latent fatalities, but early fatalities as well. (Bley/Richardson 2.1, p. 11; Thompson, Tr. 6168, 6173) For example, should a seismic event occur, differential motion between the present containment and an FVCS or SCS, which would be located some distance away due to site space availability, might cause failure of any connecting structure that links the containment to the proposed modification. Such a failure could lead to early fatalities. (Bley/Richardson 2.1, p. 18)

2.1-40 Several other failure modes for these mitigation systems involving adverse accident paths and systems failure were discussed in some detail in the

record. The parties generally agreed that the following types of failure modes were possible and would have to be considered in the design of either an FVCS or SCS:

1. Premature venting could negate the containment spray injection system (CSIS) function, which is actuated by containment overpressurization. If CSIS actuates after the FVCS has removed most of the noncondensible gases, a strong vacuum could result, leading to containment failure by implosion. (J. Meyer 2.1, p. 8; Thompson, Tr. 6167; Bley/Richardson 2.1, p. 17)

2. The operation of the mitigation device could result in hydrogen ignition in the device itself leading to its failure. (Bley/Richardson 2.1, p. 18; Thompson, Tr. 6168)

3. The vent could fail to open on demand or close when required, either negating the use of the device or placing an unacceptable load on the filters. (Bley/Richardson 2.1, pp. 18-19; Thompson, Tr. 6168)

2.1-41 The possible failure modes which would affect an SCS are very similar to those which might be encountered with an FVCS. (Thompson, Tr. 6176)

2.1-42 The occurrence of these types of adverse accident paths or systems failure could lead to health effects including early fatalities that are greater than would have otherwise been the case without the mitigative device. (Thompson, Tr. 6168-6176) Further, Staff witness Meyer testified that the possible failure modes listed above are only examples of many others which could occur. (Tr. 6839- 6840)

2.1-43 Some of the failure modes discussed by the parties would be more difficult than others to design against. For example, the failure mode associated with the creation of a vacuum, negative pressure and possible implosion of the containment could not be designed against without considerable difficulty. The reason for that difficulty is the necessary addition of a vacuum breaker which would be a major modification to the design of an FVCS or SCS. This would be a very expensive addition. (J. Meyer, Tr. 6844) Implosion of the containment would be a major threat to the public health and safety involving a significant release of radioactivity. (J. Meyer, Tr. 6871-6872)

2.1-44 A very important and difficult decision with regard to the operation of either mitigation device suggested by UCS/NYPIRG is the consideration of the pressure at which the mitigation device would be activated. In making that decision, one must consider the exact point at which the containment would fail since, if the pressure at which activation occurs is too low, there is a potential for release of fission products to the atmosphere that would not have been released if the accident did not fail the containment. If the pressure at which activation

occurs is too high, then some overpressurization events would be excluded that would not be relieved through the vent because it was activated at too late a time. (J. Meyer 2.1, p. 7; J. Meyer, Tr. 6834; Sholly, Tr. 6268-6269) One possible set point referred to by the Staff was 100 psia, which would be approximately halfway between the design and assumed failure pressure of the Indian Point containments. (J. Meyer 2.1, p. 7) However, Dr. Meyer conceded that even at this set point there might still be some accident sequences which would activate the mitigation device, but which would not have resulted in a failure of containment if there were no FVCS. This would necessarily result in releases of fission products which would not otherwise occur. (Tr. 6835) This potential for an unnecessary release of radioactivity is an attendant risk which must be considered in evaluating the advisability of installing an FVCS. (Id.)

2.1-45 The parties appear to be in agreement that with an adequate research and development program and an actual design and an associated probabilistic risk assessment, the frequency of potential failure modes and attendant risks of an FVCS or SCS can be minimized. (Bley/Richardson 2.1, p. 19; J. Meyer, Tr. 6764; Thompson, Tr. 6219) However, because of the great uncertainty con-

cerning the usefulness of previous research on FVCSs and SCSs to the Indian Point facilities, (see Finding 2.1-25 to 2.1-28, 2.1-93), this process would probably be long and costly. (Bley/ Richardson, 2.1, p. 19)

2.1-46 The Board finds that many failure modes and attendant risks of an FVCS and SCS exist, including premature venting which could increase the consequences of an accident, failure of the pressure boundary as the result of a seismic event, failure of the valves which isolate the devices from the containment, filter failure, promotion of explosive mixtures within the devices, inadvertent operation, implosion of the containment, and increase in the probability of basemat penetration. There may be other, as yet unspecified failure modes and attendant risks.

H. ACCIDENT SCENARIOS IMPACTED BY AN FVCS OR SCS

2.1-47 In order to determine the risk reduction potential for either an FVCS or SCS, it is important to first consider which types of health effects and accident sequences could be affected by the implementation of these mitigation features. In general, overpressurization

events for which a filtered vent or separate containment system might be considered can be classified into (1) early or rapid overpressurizations, e.g., hydrogen burns; (2) moderate overpressurization events, e.g., primary system blowdown and molten core quenching; and (3) slow overpressurization events, e.g., core-concrete interactions and long-term decay heat events. (J. Meyer 2.1, p. 7)

2.1-48 The parties appear to agree and the testimony indicates that a filtered vented system will be ineffective in preventing containment failure from rapid overpressurization events. The reason for this is that to accommodate rapid overpressurization events, the size of the containment penetration required would be too large. (Richardson, Tr. 6295; J. Meyer 2.1, p. 7; Thompson, Tr. 6214)

2.1-49 While UCS/NYPIRG had suggested in its direct testimony that an FVCS might offer substantial protection against hydrogen burns (a possible source of early overpressurization failure for some containment designs), that opinion was based upon a review of documents such as NUREG/CR-1410, NUREG/CR-0165, and NUREG/CR-0138, Effect of Containment Venting on the Risk from LWR Meltdown Accidents. The latter document was prepared by Battelle

Columbus Laboratories for the NRC in June 1978. (Sholly/Thompson Testimony, pp. 13-15) However, all these documents were published prior to the IPPSS, and were based upon analysis appearing in WASH-1400. (Id., pp. 16-17; Sholly, Tr. 6231, 6232, 6249; Bley/Richardson 2.1, pp. 8, 10; J. Meyer, Tr. 6714, 6853) WASH-1400 assumed the failure pressure of the containment to be approximately 100 psia, which is substantially below the failure pressure of the Indian Point containments. (Bley, Tr. 6321) Therefore, the studies relied upon by UCS/NYPIRG in offering the opinion that a filtered vented system could be useful in mitigating hydrogen burns at Indian Point are based on containment failure assumptions which are inapplicable to Indian Point. (Bley/Richardson 2.1, p. 8; J. Meyer, Tr. 6716) Given the pressure capability of the Indian Point containments, pressure surges which lead to early containment failure from pressure spikes or hydrogen explosions occurring singularly or concurrently will, for all credible scenarios, always be below 140 psia. Therefore, early containment overpressure failures are not major contributors to the risk of either early or latent health effects

at the Indian Point facilities. (Bley/Richardson 2.1, p. 6; Richardson, Tr. 6397, 6398)

2.1-50 Filtered vented containment systems could potentially be effective in preventing containment failure resulting from moderate and slow overpressurization events. (J. Meyer 2.1, p. 7) Although an FVCS can accommodate moderate overpressure incidents, the magnitude of the pressure resulting from these events is not likely to fail the Indian Point containments and thus there is no need to provide mitigation for these moderate overpressurization scenarios. (Id., p. 12; J. Meyer, Tr. 6857)

2.1-51 All of the parties agree that the FVCS, while potentially effective for overpressurization scenarios, would have essentially no effect in terms of mitigating the interfacing systems LOCA, commonly called the V sequence. This particular scenario would bypass the containment and thus would bypass the filtered vent or separate containment system as well. (J. Meyer, Tr. 6849; Bley, Tr. 6318, 6319, Richardson, Tr. 6401; Sholly, Tr. 6324, 6235) In addition, neither a containment bypass resulting from a steam generator tube rupture nor a failure to isolate the containment would be mitigated by an FVCS. (J. Meyer, Tr. 6814)

2.1-52 Filtered vented and separate containment systems do not prevent accidents nor reduce the frequency or types of accident initiators for core melt. (Bley/Richardson 2.1, p. 10) In addition, the installation of a filtered vent or separate containment system would increase the probability of containment failure by basemat penetration. (Id., p. 17; Sholly/Thompson Testimony, p. 13)

I. RISK REDUCTION POTENTIAL

2.1-53 There was substantial disagreement among the parties regarding the potential for risk reduction as a result of the installation of either a filtered vent or separate containment system. UCS/NYPIRG witnesses Sholly and Thompson relied on risk reduction calculations appearing in NUREG/CR-0165, NUREG/CR-0138, NUREG/CR-1410 and SAND/80-0887, Development and Analysis of Vent Filtered Containment Conceptual Designs (May 1980) to predict large potential reductions in all health effects and property damage from the installation of an FVCS or SCS. (Sholly/Thompson Testimony, pp. 17-18) These studies were all published prior to the IPPSS and are all based upon WASH-1400 analyses and assumptions regarding containment capa-

bility. (See Finding 2.1-49, 2.1-55 to 2.1-60) None of these documents contained Indian Point-specific risk reduction analyses for an FVCS based upon the recently determined dominant contributors to risk for the Indian Point facilities. (Bley/Richardson 2.1, p. 10; Sholly, Tr. 6249) Due to site and plant-specific analyses, the risk reduction ratios presented by these documents have diminished. (J. Meyer, Tr. 6852)

2.1-54 Since all of the studies relied upon by UCS/NYPIRG are based upon WASH-1400 analyses, they utilized source term assumptions which are conservative and unrealistic. Therefore, they would all have overstated the risk reduction potential of an FVCS. (Bernero, Tr. 12599, 12601-12602; see Stratton, et al., pp. 5-10, 64; Licensees 1, p. 22) Using source term estimates based upon recent analyses will reduce the value of mitigative devices by a factor of 10. (Stratton, et al., p. 6)

2.1-55 In NUREG/CR-0165, assuming a WASH-1400 type containment, the implementation of a filtered vented containment system was predicted to result in approximately a factor of 12 reduction in early fatality work. (Sholly/Thompson Testimony, p. 16) However, as UCS/NYPIRG conceded, when one assumes a stronger containment (defined

in NUREG/CR-0165 as one built to withstand 120 psia) which more closely resembles the Indian Point structures, the results in the report indicate a risk reduction of only 2.4. (Thompson, Tr. 6258-6259) Since the Indian Point containment has a pressure capability in excess of 120 psia, the risk reduction would be even less, using the NUREG/CR-0165 analysis. (Sholly, Tr. 6239) In addition, the risk reduction calculations presented in NUREG/CR-0165 did not take into account the effect of the V sequence, which is not mitigated by an FVCS. (Sholly, Tr. 6233, 6250) Consideration of the V sequence would have further lessened the potential risk reduction reported in that study. (Sholly, Tr. 6236)

2.1-56 NUREC/CR-0165 reported a reduction in latent fatalities from the base value of 1.0 to a value of .11 for expected latent health effects (a factor of 9). (Sholly/Thompson Testimony, p. 16) When one assumes the presence of a stronger containment as defined above (such as Indian Point), the the base case value becomes .45. (Sholly, Tr. 6258) Therefore, when one begins with the assumption of a stronger containment, the risk reduction is only a factor of 4 (from .45 to .11).

2.1-57 NUREG/CR-0165 also did not consider the attendant risks associated with these mitigation devices. (Thompson, Sholly, Tr. 6249-6251)

2.1-58 In NUREG/CR-0138, the consequences resulting from a severe accident at the Surry nuclear plant, which was used as the model for WASH-1400, were arbitrarily assigned a value of 100. The implementation of an FVCS was calculated to reduce early fatalities to 6.2% and latent fatalities to 9.8% of that base value. (Sholly/ Thompson Testimony, p. 17) However, as discussed above, NUREG/CR 0138 was based upon WASH-1400 analyses. (Sholly, Tr. 6239) Further, like NUREG/ CR-0165, the study did not consider the effect of the V sequence on the risk reduction values reported, and arbitrarily assigned a failure probability to the FVCS of 1 in 100. (Sholly/ Thompson Testimony, pp. 17-18)

2.1-59 NUREG/CR-1410 similarly relied upon WASH-1400 analyses, and did not have the benefit of the identification of dominant risk contributors for the Indian Point facilities. The study was described by Staff witness Meyer as a first pass at these mitigation features which included only a very general analysis of their practicality and feasibility. (J. Meyer, Tr. 6728) In addi-

tion, the NUREG did not analyze or quantify the attendant risks resulting from the failure modes which are possible for an FVCS. (J. Meyer, Tr. 6733, 6734, 6735) Finally, the fact that the steam spike overpressurization scenario was thought to be a dominant failure mode in NUREG/CR-1410 but is not now so viewed because of Indian Point-specific analyses, is a major factor responsible for the diminished risk reduction potential presently attributed to an FVCS. (J. Meyer, Tr. 6856, 6857)

2.1-60 SAND/80-0887 calculated the elimination of early fatalities using an FVCS based upon the occurrence of one accident sequence, the TMLB' sequence, in a four loop Westinghouse pressurized water reactor at an unspecified location "chosen because of its proximity to a population center." (Sholly/Thompson Testimony, p. 18) Once again, however, the publication date of May 1980 indicates that the analysis was based upon WASH-1400 assumptions. Since UCS/NYPIRG explicitly states that the risk curves in NUREG/CR-1410 display similar results to SAND/80-0887, (Id.), it is reasonable to assume that, like NUREG/CR-1410, SAND/80-0887 did not include an analysis or quantification of attendant risks associated with the operation of an FVCS. (See Finding 2.1-59)

2.1-61 The pre-IPPSS studies relied upon by UCS/NYPIRG could not take into account the detailed considerations which could only come from an extensive probabilistic risk assessment like the IPPSS. (J. Meyer, Tr. 6715) The risk reduction achievable for various mitigation devices is governed by the dominant contributors to risk posed by a given facility. (Thompson, Tr. 6248) The two major contributors to the risk of early fatalities at the Indian Point facilities were identified in the IPPSS as the Z-1Q sequence and the V sequence. (IPPSS, p. 8.3-1) However, the Indian Point containments have since been found to be strong enough to withstand the largest earthquakes which can reasonably be postulated for the Indian Point site. This means that the failure of the containment pressure boundary does not occur for any credible seismic event and that the Z-1Q sequence is no longer considered a possible event at Indian Point. (Bley/Richardson 2.1, pp. 4-5, 11; Bley, Tr. 6370; see Finding 1.1.-83) Thus the risk of early fatalities at Indian Point, based on the best available information, is expected to come almost entirely from the V sequence, which contributes less than one half of one percent to the overall core melt frequency. (Bley, Tr. 6377; Bley/Richardson 2.1, p. 8) Since all parties agree that neither an FVCS or SCS would be effective in mitigating

the consequences of a V sequence accident scenario, (see Finding 2.1.-51 and since the V sequence is the overwhelming contributor to the risk of early fatalities at Indian Point, neither an FVCS or SCS could possibly have a large effect on early fatalities at Indian Point. (Bley/Richardson 2.1, p. 8) Therefore, the reliance by UCS/NYPIRG on studies such as NUREG/CR-0165 and NUREG/CR-0138, which did not consider the V sequence in determining risk reduction values for early fatalities, would appear to be misplaced. (Id.)

2.1-62 An FVCS or SCS would be most effective in mitigating the consequences of a gradual containment overpressurization failure. (See Finding 2.1-50) The Licensees analysis presented under Question 1 indicates that a gradual overpressurization failure of the containment from a core melt and loss of containment heat removal capability would conservatively take twelve hours and more likely about twenty-four hours, and would result in no early fatalities. (See Finding 1.1-263; Bley/Richardson 2.1, p. 6) While the NRC Staff believes that early fatalities are possible from such a scenario under the limited circumstances of the Late-Reloc emergency response assumptions, the Staff itself characterized these assumptions as pessimistic. (See Findings 1.1-263, 1.1-368) Thus, the

NRC Staff agrees that gradual containment overpressurization failure releases seldom would result in early fatalities. (Rowsome, Tr. 12910; Rowsome/Blond 5, p. 14) Therefore, since an FVCS or SCS would not affect the V sequence, and since a gradual overpressurization failure would seldom, if ever, result in early fatalities, these mitigation devices would have little, if any, effect on reducing the early fatality risks at Indian Point. (Bley, Tr. 6282)

2.1-63 The NRC Staff did present calculations of potential factors of 2 and 4 reduction in early fatalities from Indian Point Units 2 and 3, respectively, through the implementation of an FVCS. (J. Meyer 2.1, p. 13) However, as stated in Finding 2.1-62, these results are based upon a very conservative and bounding Staff consequence analysis which calculated early fatalities from a gradual overpressurization failure of containment by making very pessimistic assumptions regarding emergency response for accidents initiated by earthquakes and hurricanes. (See Findings 1.1-263, 1.1-368) The calculations were also based upon the assumption that direct failure of the containment from a seismic event (release category Z-1Q) is possible. Elimination of the Z-1Q release category would mean that the overall risk would be reduced. (J.

Meyer, Tr. 12507-12509) The testimony does, indeed, indicate that the Z-10 release category is no longer possible for the Indian Point plants. (See Finding 1.1-83) The Staff has taken the position that, to be of value, a mitigation device must reduce risk by a factor of 10. (NUREG-0850, p. 4-2)

2.1-64 The NRC Staff calculated that a factor of five reduction in latent health effects was possible for both facilities as a result of an FVCS or SCS. (J. Meyer 2.1, p. 13) The risk reduction potential related to property damage was found to be close to that calculated for latent health effects. (J. Meyer, Tr. 6813) The potential for risk reduction is less than the factor of 10 which the Staff believes to be the threshold for mitigative features to be of value. (See Finding 2.1-63) The potential for a larger reduction in latent health effects than in early health effects resulted from the Staff's finding that 95% of the risk of latent effects at Indian Point are due to the gradual overpressurization failures for which the FVCS and SCS are most effective. (J. Meyer, Tr. 6742-6743) The NRC Staff also testified that the incidence of nuclear accident caused cancers would probably be indistinguishable from non-nuclear cancers, that is, a fairly small percentage of the background cancer incidents. (Rowsome, Tr. 12927)

2.1-65 Dr. Meyer emphasized that the calculated values for risk reduction potential appearing in the Staff's testimony are only achievable if the mitigation features employed are qualified to withstand the external events which are the dominant contributors to risk at Indian Point. Risk reduction potential for either an FVCS or SCS would be negligible unless they are qualified for these key external events. (J. Meyer 2.1, pp. 3, 19-20; J. Meyer, Tr. 6813-6814)

2.1-66 Dr. Meyer's estimates of risk reduction potential, based upon consequence modeling utilizing overly conservative WASH-1400 source terms, overstate the potential benefits of an FVCS or SCS. (See Bernero, Tr. 12599)

2.1-67 Regarding the differences in the estimates of potential risk reduction for these mitigation devices contained in the NUREG documents relied upon by UCS/NYPIRG compared to the much smaller estimates contained in the Staff's testimony, the lower estimates of risk reduction potential result from taking into account in the Staff's testimony site-specific and plant-specific features not taken into account elsewhere. The Staff's

current calculations are also based upon the modifications in equipment and procedures made at the facilities recently to reduce the seismic, hurricane and fire risks. (J. Meyer, Tr. 6852, 6745, 6787) Further, the Staff's current assessment took into account the attendant risks associated with an FVCS and SCS. (J. Meyer 2.1, p. 9)

2.1-68 Licensees agree that either of the mitigation devices suggested by UCS/NYPIRG could potentially have a positive impact on latent health effects and property damage resulting from a gradual overpressurization failure of the Indian Point containments. (Bley/Richardson 2.1, p. 10) However, the Licensees believe that any calculation of the potential risk reduction for latent effect and property damage as a result of the implementation of these mitigation features would be premature at this time. The Licensees testified that without a risk analysis of the proposed mitigation features joined to and interacting with the Indian Point containments, a risk reduction value attributable to the proposed features cannot be established. (Bley/Richardson 2.1, p. 19) Further, the Licensees testified that a precise risk reduction value can only be calculated following the completion of the detailed study of source terms that is now ongoing at the NRC and in the industry. (Id., p. 21)

2.1-69 The containment pressure capability of 141 psia for the Indian Point plants, (see Finding 1.1-408) which results in a delay in containment overpressurization of at least 12-24 hours following accident initiation, makes these devices even less attractive because the time available prior to containment failure permits the opportunity for safety system recovery and decay of short-lived isotopes, as well as evacuation if necessary. Failure probability of the containments was conservatively estimated by the Licensees in the IPPSS because recovery actions following core degradation were not quantified. (Bley/Richardson 2.1, pp. 6-7)

2.1-70 Both the Licensees and the NRC Staff agree that if more realistic source terms were applied to the analysis, the risk reduction potential of either an FVCS or SCS would be correspondingly reduced and the incentive for backfits to lower risk would be diminished. (Bley/Richardson 2.1, p. 21; Rowsome, Blond, Tr. 12914, 12915)

J. COST

2.1-71 The true cost of either an FVCS or SCS can only be accurately estimated after completing a rather

extensive site and plant-specific design analysis. (J. Meyer, Tr. 6786) The best approach to preparing such an analysis is a probabilistic risk assessment on a specific design, which would have a large impact on cost. (Bley/Richardson 2.1, p. 11)

2.1-72 The detailed effort which would be required to perform a plant-specific design analysis leading to accurate cost estimates for these mitigation features would include the following methodological components: (1) the Indian Point plant-specific accident initiation frequencies, (2) the inclusion of a specific study of the effect of operation, misoperation or failure of the mitigation system on the probability of reaching different degraded core situations, (3) the effect of the mitigation system operation or misoperation on the containment failure probability, (4) the effect of the mitigation system operation or misoperation on the timing of and type of radiological release, and (5) the inclusion of the above items in a detailed assessment of the probability and consequences of core meltdown accidents at Indian Point. (Bley/Richardson 2.1, pp. 12-13) The study would also have to include the design considerations which have been previously discussed. (See Finding 2.1-30)

2.1-73 Although no precise cost estimates can be done without a detailed analysis, both UCS/NYPIRG and the NRC Staff presented "ball park" estimates of the cost of these features based on the available literature. NUREG/CR-1410, for example, estimated the cost of an FVCS at approximately $12-$32 million depending upon the seismic requirements and filtering capability. (Sholly/Thompson Testimony, p. 19; J. Meyer 2.1, p. 9) The estimate of $12-$32 million does not include qualification for external events, reactor down time, licensing delays or special instrumentation inside the containment. (J. Meyer 2.1, p. 13; J. Meyer, Tr. 6695, 6785, 6786; Sholly/Thompson Testimony, p. 18; Thompson, Tr. 6186) UCS/NYPIRG estimated that as an upperbound figure, the cost of these mitigation features would not exceed $100,000 million. (Sholly/Thompson Testimony, p. 20) UCS/NYPIRG witness Thompson expressed the view that the reactor down-time component of the cost of these mitigation devices would be minimal assuming no additional containment penetrations were necessary. In that event, Dr. Thompson believes that the construction of the mitigation features could be done during a refueling outage in a matter of weeks. (Thompson, Tr. 6186, 6187) However, Dr. Thompson conceded that if additional penetrations were necessary, existing systems would have to be modified, and the construction time would

be substantially greater. (Tr. 6187) Dr. Thompson did not have an opinion as to whether an additional penetration would be necessary for installation of either of these mitigation features at Indian Point. (Thompson, Tr. 6188, 6221)

2.1-74 The $100 million upper bound cost estimate offered by UCS/NYPIRG does not include the cost of research and development for these mitigation devices. (Thompson, Tr. 6186) In addition, neither the cost of lost electrical generation during construction and technical specification outages, (Thompson, Tr. 6186), nor the cost of operating and maintaining the FVCS, (Thompson, Tr. 6219), were included. Dr. Thompson testified that the cost of research and development could be in the low or high tens of millions of dollars for an FVCS depending upon the adaptability of the Swedish work done on the Barseback boiling water reactor to Indian Point. (Tr. 6191-6192) UCS/NYPIRG has done no analysis to determine the applicability of the Barseback design to large dry containments such as those at Indian Point. (Id.)

2.1-75 With specific regard to a separate containment system, the cost of such a device would be high since a new building with equivalent capability to the present containment to withstand initiating events would

have to be constructed. The cost estimate offered by the Staff was approximately $100 million per structure. (J. Meyer 2.1, pp. 15-16) Since it appears that Unit 1 would need seismic upgrading in order to serve as a separate containment system at Indian Point, (see Finding 2.1-36), the cost of adapting that structure for use as a separate containment system would be greater than or equal to the cost of an FVCS. (Id., p. 20)

2.1-76 NRC Staff estimates of the value of mitigation provided by a controlled, filtered vent or an add-on, fully qualified separate containment structure make it clear that these concepts are not cost-effective for reducing risk, that is, they would cost substantially more than the risk reduction value would warrant. (Rowsome/Blond 5, p. C-32 to C-33)

2.1-77 Another way to assess the value of mitigation is in terms of cost per fatality averted. (See B. Cohen 5, pp. 16-25) Assuming a total cost of 50 million dollars for designing and installing an FVCS and using the IPPSS estimate of 1.7×10^{-1} per reactor year expected latent fatalities at Indian Point Unit 2, the cost per fatality averted (in 1975 dollars) associated with the installation of a FVCS is 10.9 million dollars. (B. Cohen 5, p. 25) This compares with 320,000 dollars per fatality

averted for the installation of air bags in automobiles, 60,000 dollars per fatality averted for the installation of home smoke alarms, and 30,000 dollars per fatality averted for further medica' screening and care for breast cancer. (Id., p. 27)

K. OTHER MITIGATION FEATURES

2.1-78 In addition to a filtered-vented containment system and an add-on separate containment building, the NRC Staff also evaluated other mitigation features. The mitigation strategy considered by the NRC Staff consisted of a triad of features: glow plug igniters to control combustible gases; a passive containment building heat removal system consisting of heat pipes to control overpressurization; and a system to insure the flooding of the reactor cavity to prevent basemat penetration. (Meyer/Pratt Testimony, p. III.B-29) The cost of these features would likely be several tens of millions of dollars. (Rowsome/Blond 5, p. C-33)

2.1-79 Glow plug ignitors would mitigate the hydrogen burn containment failure mode by providing for the controlled burning of an amount of combustible gas sufficient to render the containment building inert by oxygen depletion in such a way that thermal or pressure

loadings from controlled burning do not cause failure of vital equipment or the containment building. (Meyer/Pratt Testimony, p. III.B-28) Two possible attendant risks of glow plugs include an increase in containment pressure and some potential adverse equipment burns. (J. Meyer, Tr. 12576) In vessel failure situations in which large amounts of hydrogen would be released at one time, a glow plug could have an attendant risk of igniting a relatively rich mixture of hydrogen. Without this device, the mixture could lean out by mixing with the rest of the atmosphere and become less of a hazard. (J. Meyer, Tr. 12578-12579)

2.1-80 A heat pipe system is a closed pipe system which contains a working fluid. The portion of the pipe inside containment evaporates the fluid and thus removes heat from containment by condensation of water vapors on the outside surfaces of the pipe. Heat from the containment is eventually released to the atmosphere. (J. Meyer, Tr. 12568-12569) No heat pipe system has ever been tested in any prototype or design demonstration program for any nuclear power plant. (J. Meyer, Tr. 12569) The wall thickness of the pipes in the study of heat pipes upon which the Staff analysis is based (NUREG/CR-1666) is three millimeters. The devices would have to be qualified for internal and external events. There would be engine-

ering challenges to produce such a system for practical application to the Indian Point containment design. Design and implementation would be extremely costly. (J. Meyer, Tr. 12570-12571) Another major consideration would be the possibility of having to cut through layers of rebar to create a new containment penetration for the heat pipes. (J. Meyer, Tr. 12572-12573)

2.1-81 Flooding the reactor cavity as suggested by the Staff would limit core/concrete interactions by establishing a coolable debris bed in the reactor cavity, and would thereby control basemat penetration. (Meyer/Pratt Testimony, p. III.B-29) However, the requirement for a flooded cavity would be counterproductive if it is not installed in conjunction with heat pipes. (J. Meyer, Tr. 12573) Without providing for containment heat removal, the cavity flooding system would actually increase risk because of the increased potential for containment overpressurization failure. (Meyer/Pratt Testimony, p. III.B-68)

2.1-82 Of the release categories that dominate risk, only the slow containment overpressurization scenario would be reduced or eliminated by the mitigation strategy. (Meyer/Pratt Testimony, p. III.B-40). According to the Staff, the risk reduction in delayed or latent cancers

which might be expected is only a factor of 5 assuming a "realistic" view of the mitigation strategy, that is, attempting to account for negative characteristics and unreliability of each feature in the strategy. (Meyer/ Pratt Testimony, p. III. B-45)

2.1-83 The Staff estimated the economic worth of Dr. Meyer's triad strategy for Indian Point using four different mathematical algorithms. These algorithms provide a means of estimating the dollar value of risk reduction. The Staff's estimates of present worth of the risk reduction strategy (i.e., the dollar value of health effects averted) ranged from 3.3 million to 20.8 million dollars. (Rowsome/Blond 5, p. C-32) This range of values is the basis for the Staff's comparison with the actual cost of the mitigative features. (Id., pp. C-32 to C-33) If it were possible to implement Dr. Meyer's mitigation features for 10 million dollars, it would still be unclear whether the fix were cost-effective. However, it is likely that Dr. Meyer's suggested strategy would cost "several tens of millions of dollars," and the Staff does not recommend its implementation. (Id.)

2.1-84 Additionally, in the 1980 Staff Task Action Plan, heavy emphasis had been placed on mitigation factors such as hydrogen control, controlled filtered vent-

ing of containment, and a core retention device because it was not clear at that time that the plants as built provided effective mitigation of the offsite radiological consequences of core melt accidents. Subsequent study indicated that gradual overpressurization failure of containment will not take place with an operating containment heat removal system, will take a long time to develop in any case, and will almost never result in early fatalities even in the absence of evacuation. Basemat melt through has been found to produce very little offsite radiological risk. Thus most of the desirable attributes of the early mitigation backfit concepts are already present in the plants. (Rowsome/Blond 5, pp. C-13 to C-14)

2.1-85 Because of the above and the fact that Indian Point poses no undue risk, the NRC Staff recommends that the NRC not issue orders to implement backfits to improve the mitigation of severe reactor accidents. (Rowsome/Blond 5, p. C-13)

2.1-86 The NRC Staff does not recommend that a high-priority, plant-specific program to search out and require modest benefit, low-cost design improvements be implemented because the risk of the Indian Point plants is small and within the range of risks posed by other plants licensed to operate by the Commission. Further, on-going

generic activities for all operating plants, such as the Severe Accident Policy Development Program, provide a more appropriate venue for pursuing such design improvements. (Rowsome/Blond 5, pp. C-12 to C-13.)

2.1-87 The NRC Staff did recommend that the Licensees be ordered to implement a Safety Assurance Program (SAP), to explore further risk reduction. The object of such a program is to assure that the conduct of operations and future clues to the safety of the plants are and remain consistent with the level of risk assessed by the Staff. (Rowsome/Blond 5, p. C-16)

2.1-88 The Staff feels it to be highly desirable that the program be developed by the Licensees rather than prescriptively imposed upon them and that the Licensees integrate the program into their operations organization and minimize the extent to which it is an external or contracted function. The specifics of the program would be developed subject to the advice and consent of the Staff. (Rowsome/Blond 5, p. C-19)

2.1-89 The Staff anticipated the program startup costs to be a few million dollars with continuing costs of a few hundred thousand dollars per year. Although the residual risk is not large, Staff concluded that the com-

paratively low cost associated with studies and alteration to procedures or training suggest that such a program is likely to be cost-effective. (Rowsome/Blond 5, p. C-17)

2.1-90 The Staff did not perform a monetization benefit calculation for the Safety Assurance Program. (Rowsome, Tr. 12929) There was neither an intensive evaluation of the effectiveness of such a program nor a review to see whether existing programs, activities or procedures already in place at the plants would address the same purposes as the Safety Assurance Program. (Rowsome, Tr. 12931-12932, 12934-12937)

2.1-91 Incentive for such an assurance program would be diminished with lower source terms. (Rowsome, Tr. 12938)

2.1-92 Staff witness Rowsome testified that he and a substantial community within the Staff believe the Safety Assurance Program ought to be in fact generic. This belief is based on the same rationale used to determine that backfits are not required and that the search for further plant improvements to reduce risk should be folded into generic activities for all operating plants. That rationale is that the risk of the Indian Point plants is low and within the range of risks of other

plants licensed to operate by the Commission. (Rowsome, Tr. 12933-12934)

L. ULTIMATE FINDINGS

2.1-93 Based on the direct testimony and extensive cross-examination conducted on Contentions 2.1(a) and (d), the Board finds that most of the previous work which has been done with filtered vented and separate containment systems, e.g., at the ZZPR, the FFTF, Barseback, etc. is of questionable applicability to any specific design for implementation of these mitigation features at Indian Point. The facilities which are the subject of the prior design work on these mitigation features are too dissimilar to the Indian Point facilities to provide substantial confidence that a significant portion of the work already completed could be adapted for use at Indian Point. Nevertheless, the Board finds that given a proper research and development program, the devices could be made feasible for use at Indian Point. However, it appears that the detailed site-specific and plant-specific probabilistic risk assessment study which would be required prior to making a decision regarding implementation of these mitigation features would take a significant period of time to complete and would be very costly.

2.1-94 With specific regard to the suggestion that Indian Point Unit 1 be used as a separate containment system, the Board finds that although the use of that structure might be technically feasible, the modifications that might be necessary in order to so use that facility would be very costly and would likely take a significant period of time to accomplish.

2.1-95 Further, the Board finds that when one takes into account the cost for qualification of these mitigation features for external events, possible special instrumentation inside the containment, reactor down time, licensing delays, and research and development, the potential cost of either mitigation feature could well exceed $100 million. In order to justify such a significant expenditure, the Board believes that the risk reduction potential of a mitigation device should be substantial. We do not find that such is the case with either an FVCS or SCS. The risk reduction potential for devices to mitigate gradual overpressurization failures as supported by UCS/NYPIRG in this proceeding is based upon documents which are now outdated and source term assumptions which are overly conservative.

2.1-96 Since the Staff's calculation of risk reduction of a factor of five for latent health effects

and a factor of two for early health effects for the FVCS, SCS or triad mitigation strategy are based upon a consequence analysis which we have previously found to be conservative and bounding, (see Findings 1.1-364), and source terms which are overly conservative, (see Finding 1.1-409)the Board finds that the potential risk reduction from the use of an FVCS, SCS or any of the other proposed mitiga- tion devices could well be below those predicted by the Staff. With particular regard to early fatalities, the Board finds that the implementation of any of the mitiga- tion features would have little if any effect on early fatalities as a consequence of a core melt accident at Indian Point. In any case, we agree with the Licensees that the determination of the true risk reduction potential for any specific design would have to follow the completion of a detailed site and plant specific analysis.

2.1-97 The Board is in agreement with the conclusion of the Staff's monetized benefit evaluation that no further design changes at Indian Point would be cost-effective in reducing risk.

2.1-98 Additionally, the Board has previously found that, with respect to the first prong of the Commission's two-prong test, there is no significant risk associated with the operation of the Indian Point facilities.

This is especially so since the facilities meet both the proposed individual and societal public health safety goals of the Commission, even without additional safety features. (See Findings 5.0-6 to 5.0-11) Even if the operation of these facilities did pose such a risk, there is no basis in this record for the Board to find that a significant reduction in the risk posed by these facilities is possible through the implementation of either an FVCS, SCS or any other proposed mitigation features or that such risk reduction would be cost effective. Therefore, the Board finds that the evidence does not justify the need to implement either an FVCS or SCS or any other mitigation device as a condition to the continued operation of the Indian Point facilities.

2.1-99 The Board also agrees with the NRC Staff that a high priority, plant-specific program to search out and require modest-benefit, low cost design improvements to further reduce risk need not be implemented. We believe, as stated by the Staff, that since the risk posed by the Indian Point plants is low and within the range of risk posed by other licensed nuclear power plants, matters relating to mitigation and risk reduction would best be folded into the generic activities for all operating reactors.

2.1-100 For similar reasons, the Board finds that a Safety Assurance Program should not be required of the Licensees. No compelling reasons have been offered for singling out the Indian Point plants for such a program. Further, no evaluation of the effectiveness of such a program has been performed, nor has any review been conducted of measures already in effect at the plants which may serve the same purposes as the proposed program. We believe that the elements of this program are not sufficiently mature to be imposed on a single Licensee and that, as suggested by Staff witness Rowsome, the program should be developed on a generic basis. In fact, the need to implement such a program at all may be obviated by upcoming developments in source term technology.

V. CONTENTION 2.2(a)

2.2-1 Contention 2.2(a) reads as follows:

> The cooling system at the plants should be changed so that it no longer uses brackish Hudson River water. This change is needed to combat safety-related corrosion problems.

2.2-2 WBCA maintained in its testimony that leaks in fan cooler units are caused by the use of brackish water. (Fleisher 2.2(a), pp. 4-5; Fleisher, Tr. 6478) Mr. Fleisher testified that the risk of brackish water is its corrosive qualities, and that there is no solution for the problem of corrosion inside containment

other than the removal of brackish water from inside the containment building. (Fleisher 2.2(a), p. 6; Tr. 6478)

2.2-3 The only specific alleged risks which WBCA was able to identify were the potential wetting of the reactor vessel with cold water as a result of leakage inside containment, and the potential wetting and subsequent shorting out of electrical circuits as a result of the same type of leakage. (Fleisher, Tr. 6479) However, WBCA witness Fleisher testified that he had done no analysis of the risk posed by vessel wetting. (Tr. 6480) Nor was he aware of any study of the effects of wetting electrical equipment regarding the risk posed by the use of brackish water. (Tr. 6481)

2.2-4 After investigating various alternative cooling systems, Con Edison did develop a detailed design of a hybrid closed cooling system, but concluded that, although possessing the advantages of a closed system during normal plant operation, the hybrid system is complex in comparison with the present once-through cooling system. (Rothstein/Tuthill Testimony, p. 7) Such a hybrid system would require a heat exchanger inside containment, such as now exists, but would also require another heat exchanger outside of containment to cool the water as it circulates in the closed loop, thereby introducing what

could be termed an additional system. During the open-loop portion of operation of the hybrid system it would be necessary to switch over from the closed loop to the once-through river cooling that now exists. This would require a system of valving which is not necessary at present. (Rothstein, Tr. 6532)

2.2-5 The cooling system preferred by WBCA is a closed cycle system using highly purified water within containment isolated by a heat exchanger from the external cooling source. (Fleisher 2.2(a), pp. 6-7; Fleisher, Tr. 6483) Although WBCA maintained that a closed cycle system was necessary to eliminate corrosion inside containment at the Indian Point facilities, its witness had done no feasibility study related to the implementation of his preferred system at Indian Point. (Fleisher, Tr. 6483)

A. EFFECT OF BRACKISH WATER ON COOLING SYSTEM MATERIALS

2.2-6 There is nothing unusual about the use of brackish or saline water for cooling power plants. Fifteen percent of domestic power plants and 33% of domestic nuclear plants are on brackish estuaries or sea coast locations and saline water is routinely used for cooling purposes. (Rothstein/Tuthill Testimony, pp. 4-5)

2.2-7 The Indian Point Unit 2 facility has five fan cooler units. Each unit consists of a main heat exchanger formed by separate heat exchanger units which cool the air, a fan to draw the air through the heat exchangers and circulate it throughout the containment, a demister to remove condensed moisture droplets from the air, and various filters to clean the air. In addition, each fan motor is cooled by a separate smaller heat exchanger referred to as a motor cooler. (Rothstein/ Tuthill Testimony, p. 2)

2.2-8 The fan cooler system is supplied with cooling water from the essential service water system through two 24 inch diameter cement-lined carbon steel pipes. Each fan cooler unit is then serviced by a separate 10 inch cement lined supply and discharge pipe. (Rothstein/Tuthill Testimony, pp. 2-3) The fan cooler U-tubes, tube sheets, water boxes and inlet piping are all made from a copper/nickel alloy. (Id.)

2.2-9 Hudson River water at Indian Point is of low salinity and might well be classified as fresh water for most of the year. (Rothstein/Tuthill Testimony, pp. 2, 3, 6) In addition, the other characteristics of the Hudson River water at Indian Point (low Ph level, low

biological oxygen demand, and dissolved oxygen which approaches saturation) are all indicative of water in which copper alloys exhibit very low corrosion rates. (Id., pp. 3-4) These low corrosion rates translate into lifetimes of 40 years for 18 gauge copper/nickel tubing, the standard gauge used in the fan and motor coolers at Indian Point Unit 2. (Id., p. 4) The copper/nickel alloy used in the Unit 2 fan coolers is the usual choice for naval and commercial applications and for use in power and industrial plants using saline water for cooling. (Id., p. 7)

2.2-10 Carbon steel such as that used in the piping which services the fan coolers corrodes at a low but significant rate in some saline waters. For this reason carbon steel piping is normally cement or epoxy-lined for brackish or saline water service, just as was done at Indian Point Unit 2. (Rothstein/Tuthill Testimony, p. 4)

2.2.-11 Hudson River water is not corrosive to cement-lined steel or copper base alloys such as those used at Indian Point 2 at flow velocities typical of Unit 2 operation. (Id., p. 6)

B. CORROSION - CAUSES AND CORRECTIVE MEASURES

2.2.-12 There are three types of corrosion which apparently caused leakages in the Indian Point 2 fan cooler units in 1980-81. First, the fan cooler coils at that time were built out of several short sections, the joints of which were brazed. (Rothstein, Tr. 6518) These brazed joints were much less than perfect and exhibited voids and/or porosity. These voids and/or porosity resulted in corrosion and leakage from the joints. (Rothstein, Tr. 6526)

2.2-13 The second type of corrosion was a form of pitting resulting from deposition of silt on the inside surface of the fan cooler tubing. Pitting developed under the silt because the water composition was slightly different under the silt from the water surrounding the silt. Very localized but significant corrosion resulted. (Rothstein, Tr. 6526 -6527)

2.2-14 The third type of corrosion which contributed to the leakage was that which occurred at welded joints of the coated carbon steel pipe which supplies the fan coolers. The concrete coating had spalled off at the welded joints, thereby exposing the bare carbon steel to the service water. (Rothstein, Tr. 6528-6529)

2.2-15 Con Edison's testimony established that the particular types of corrosion which occurred in the fan coolers prior to their replacement were not associated with either the use of brackish water or the types of materials used in the manufacture of the fan coolers or associated piping. (Rothstein, Tr. 6521)

2.2-16 Con Edison has made significant modifications which eliminate the sources of corrosion which caused the 1980 leakage. First, following the incident, all fan cooler coils, header and branch piping were replaced at Indian Point Unit 2. (Rothstein, Tr. 6517) All brazed joints were removed from fan cooler tubes and each tube is now a continuous length bent into a U shape so there is essentially one-piece construction. (Rothstein, Tr. 6518, 6520) In addition, the design of the fan coolers was changed so as to provide for conventional water boxes which can be opened when the coolers are not in service, making it possible to examine the inside of the water boxes and the tubes. (Rothstein, Tr. 6519; LeFave, et al., p. 11)

2.2-17 During the recent refueling outage, maintenance improvements were implemented on the recommen-

dation of Con Edison witness Tuthill. Examination of the copper/nickel piping, waterboxes and tubesheets in the fan cooler units demonstrated that they were in good condition consistent with the views of both Con Edison witnesses on the low corrosivity of Hudson River water. (Rothstein/ Tuthill Testimony, pp. 5-6)

2.2-18 To eliminate the pitting which was caused by silt deposition, the flow rate through the fan cooler coils has been increased from one foot per second to four feet per second. At three to four feet per second silt does not deposit on the fan cooler tubes. (Rothstein, Tr. 6527-6528)

2.2-19 In order to eliminate the corrosion which had taken place at the welded joints of the carbon steel piping, each welded joint was examined during the 1980-81 outage, and where the cement coating was incomplete, the surface of the carbon steel was cleaned, built up as necessary with weld deposit, and the inside surface was coated with cement. (Rothstein, Tr. 6529) During the refueling and maintenance outage which ended in December 1982, the welded joints in the carbon steel piping were reexamined. In every case the cement coating was still intact and no corrosion had taken place. (Id.)

2.2-20 In order to prevent recurrence of fan cooler leakage, many other modifications have been made in equipment and procedures. For example, the new fan cooler units allow access for tube repairs and tube plugging, and service water isolation valves to the fan cooler units have been replaced with an improved design. (LeFave, et al., p. 11) The inspection of the welds in the cement line carbon steel pipe during the 1982 refueling and maintenance outage was made possible by the installation of inspection ports adjacent to the welds. These not only permitted repairs of the previous weld and cement coating, but will permit inspection during future outages to maintain surveillance on the condition of the cement lining. (Rothstein/Tuthill Testimony, p. 6)

2.2-21 Other modifications include the installation of equipment in the control room and in the containment sump which will achieve improved level control and indication for the containment sump and sump pumps. The reactor cavity pump control and CCR cavity level indication have been upgraded. In addition, there is continucus sump level indication in the control room. Finally, leak detection equipment and procedures for the containment building have been upgraded and there are now

revised technical specifications regarding limiting conditions for operation for the various leakage detection systems and for operating leakage limits. (Bley 2.2(a), p. 2; LeFave, et al., pp. 8, 11, 13)

2.2-22 As a result of the modifications made at Unit 2, while some leakage may still occur inside containment, such leakage will be detected and controlled, which reduces the probability of damage to safety-related equipment and decreases the frequency of occurrence of flooding inside the containment building. (Bley 2.2(a), p. 2; Bley, Tr. 6428, 6430) For example, the Staff testified that although there have been seven or eight incidents of leakage since the fan cooler repairs and modifications in 1980, all the leaks have been discovered and repaired before the technical specification limits were exceeded. (LeFave, Tr. 6904)

2.2-23 In any case, the containment systems are designed for flooding under the design basis accident, which is a large LOCA in which substantial amounts of water made conductive through the addition of boric acid (well in excess of 300,000 gallons under accident conditions) is released into containment. The safety equipment that needs to function inside containment is

designed with the expectation that a significant amount of conductive water will be present. Therefore, under normal conditions a tiny fraction of the amount of water assumed for the design basis accident will not threaten any of the safety functions. (Rowsome, Tr. 6905)

2.2-24 Even though significant steps have been taken to prevent recurrence of significant leakage at Unit 2, these modifications in equipment and procedures have affected primarily the reliability and longevity of the fan coolers and associated systems inside containment. Both the Staff and the Licensees testified that the brackish water issue is primarily a maintenance (economic) concern, with minimal implications for health effects risks, even in the unlikely event that containment fan cooler leakage were to go undetected for a long period of time. (Bley 2.2(a), p. 1; LeFave, et al., pp. 3, 8; Rowsome, Tr. 6891) Prior instances of fan cooler leakage were not significant from a public health and safety standpoint. (Bley, Tr. 6428)

2.2-25 WBCA's proposed change in the cooling system at Indian Point would not have a significant effect on the risk of core melt. (Holahan/Rowsome Testimony, p. 13) There is very little chance of safety-related effects

from leakage of water, brackish or not, into the containment. (Bley, Tr. 6429; Turovlin, Tr. 6912) Failures of the fan cooler units were included in the risk analyses of both the Staff and Licensees. These were found to have very little effect on risk, since risk is dominated by events which defeat all active safety systems at once. (LeFave, et al., pp. 7-8) The metallurgical and design improvements in the fan cooler units of Indian Point Unit 2 would further reduce the risk associated with the potential corrosive properties of brackish water. (Bley 2.2(a), p. 3)

2.2-26 The Staff testified that leakage in the containment fan coolers does not warrant the requirements suggested by WBCA of a closed loop design for the Indian Point facilities. Such a design change would only be required by the Staff if safe shut down would be precluded in the event of a corrosion-related failure. Due to the implementation of the modified detection equipment, new surveillance requirements and plant technical specifications described by the Staff and the Licensees, there is little concern that a sudden, catastrophic failure of the service water system due to corrosion could affect plant safety systems or interfere with safe plant shut down. Similarly, these modifications would prevent a small leak

from developing into a significantly large leak. (LeFave, et al., p. 15)

C. ULTIMATE FINDING

2.2-27 Based upon the testimony offered by the parties regarding this contention, the Board finds that fan cooler corrosion which occurred at Indian Point Unit 2 prior to the substantial systems modifications performed in 1980-81 was not caused by the use of brackish water as a cooling medium inside containment, and did not result in any increase in risk to the public. The equipment and operating procedure modifications which have been made by Con Edison since unit replacement in 1980-81 will assure that the incidence of water leakage will be kept to a minimum in the future and will be detected before it reaches major proportions. In any case, since leaks, even if substantial, would not compromise the ability of the plant to achieve safe shutdown, no risk to the public is presented by the continued use of the current once-through cooling system. The Board finds therefore that no significant reduction in the risk posed by operation of the Indian Point plants would be achieved by elimination of the use of brackish water as suggested by Contention 2.2(a).

VI. BOARD QUESTION 2.2.1

2.2.1-1 Board Question 2.2.1 reads as follows

> Should any of the requirements proposed at the July 29, 1982 meeting of the NRC Staff and members of the SCOG be required for Indian Point Units 2 and/or 3, considering the risk of a steam generator tube rupture in this high population area?

2.2.1-2 In a pressurized water reactor such as Indian Point Unit 2, the steam generators isolate the radioactive reactor coolant from the nonradioactive steam cycle, and serve as the heat excnanger between the two systems. At Indian Point Unit 2 there are four Westinghouse Series "44" steam generators, each of which has 3,260 tubes and six tube support plates. Indian Point Unit 2 has been operating since 1973 and has operated for approximately 38,000 full power hours. In the United States there are approximately thirty steam generators of the same model as those used at Unit 2 which have collectively accumulated over one half million full power hours of operation. (Rothstein Testimony, pp. 2-3)

2.2.1-3 The concern over steam generator tube degradation stems from the fact that the steam generator tubes are a part of the reactor coolant system boundary and that tube failures result in a loss of primary cool-

ant. Steam generator tube failure would allow primary coolant into the secondary side of the steam generators increasing the risk of release of radioactive material to the environment. In addition, steam generator tube ruptures could have the effect of compromising the availability of cooling water to the reactor core. (Holahan/Rowsome Testimony, pp. 2-3)

A. STEAM GENERATOR TUBE FAILURES - CONTRIBUTION TO RISK

2.2.1-4 The Staff performed a generic analysis of steam generator tube failures leading to core melt and concluded that the total core melt probability from all causes involving steam generator tube failures is estimated to be 4×10^{-6} per reactor year. In performing this analysis fifteen different accidents sequences were studied. The sequences address single and multiple tube ruptures, steam line failures, steam generator safety valve failures, large loss of coolant accidents, loss of auxiliary feed water, loss of RHR, Anticipated Transients Without Scram (ATWS) and various operator errors. (Holahan/Rowsome Testimony, p. 4) The generic analysis was examined by the Staff and found to be applicable to the Indian Point units. For the generic analysis, the largest contributors

to risk were found to be multiple tube ruptures combined with failure of a steam generator safety valve and an ATWS resulting in multiple tube failures. For these scenarios, the important parameters to examine are the multiple tube rupture probability, the ECCS capacity, the refueling water storage tank size, the steam generator size, and safety valve failure probability. These parameters were checked by the Staff and found to be applicable to the Indian Point units. Although no steam generator tube rupture has ever occurred at either Indian Point facility, the steam generator design and operating experience are typical of other plants. (Id., pp. 5-6)

2.2.1-5 Based on the generic analysis and its verification for the Indian Point facilities, the Staff estimated the contribution of steam generator tube ruptures leading to core melt to be approximately one percent of the total off-site radiological risk. (Holahan/Rowsome Testimony, pp. 7-8) This estimate of risk is based on a conservative Staff estimate of the radiation release attributable to these events. That is, the release to be expected of steam generator tube rupture scenarios is roughly analogous to that expected of a core melt with release to the containment with containment cooling available but an open containment penetration. This is

similar to what one would expect of a "leaking containment" model of a core melt accident. Nothwithstanding that expectation, for purposes of assessing the risk from operation of the plants under Commission Question 1, the Staff has utilized for steam generator tube ruptures, the release category developed for late overpressure failure of the containment rather than the containment leakage release category. This is a conservative assumption in the sense that it would overstate the risk caused by a tube rupture sequence by a factor of 10 or more. Such an assumption for purposes of the analysis provides confidence that the contribution to risk posed by steam generator tube rupture has not been underestimated. (Id., pp. 10-11)

2.2.1-6 Additionally, Westinghouse has analyzed a postulated event consisting of a simultaneous steam generator tube rupture and a large loss of coolant accident for a typical Westinghouse plant. The results, which would be applicable to Indian Point Unit 2, showed a small increase (less than 10° F) in the peak fuel cladding temperature when a leak rate consistent with a double-ended break of a steam generator tube (250 gpm) was assumed. Therefore, a large number of steam generator tubes would have to rupture before the peak fuel clad temperature

would approach the limit of 10 CFR 50, Appendix K. (Richardson, Tr. 12703-12705)

2.2.1-7 Since the risk attributable to steam generator tube ruptures is so small in comparison with other core melt risks at the Indian Point units, the adoption of the proposed generic requirements of the SGOG, (see Rothstein Testimony, Attachment SR-2), could at best afford only a very small risk reduction and would not alter the conclusion of both the Licensees and NRC Staff analyses (as developed in the IPPSS and Staff) that steam generator tube rupture events are not substantial contributors to risk at the Indian Point facilities. (Bley/Richardson 2.2.1, p. 6; Holahan/Rowsome Testimony, p. 11; Holahan, Tr. 6920 - 6922)

2.2.1-8 The scenario of a steam generator tube rupture coincident with a stuck open valve has been analyzed by the Licensees. It was determined that this scenario is not a major contributor to risk. The Licensees also analyzed the release of a large quantity of reactor coolant, considering the highest allowable limits on activity under the technical specifications. This scenario does not show up on the risk curves, and it is several orders of magnitude below any of the curves which are shown in the IPPSS. (Bley, Richardson, Tr. 6446-6448)

B. CURRENT CONDITION OF INDIAN POINT UNIT 2 STEAM GENERATORS

2.2.1-9 From a system reliability standpoint, the minimizing of corrosion is important because corrosion deposits fill the clearance space in the annuli between the tubes and the tube support plates in the steam generators, and can exert compressive forces on the tubing and nonuniform in-plane loadings on the remaining support plate materials. This results in deformation of the tubes (denting), and in-plane expansion and distortion of the support plates. In extreme cases the "denting" may result in tube cracking in the region of maximum tube distortion. (Rothstein Testimony, pp. 4-5)

2.2.1-10 In order to prevent the occurrence of the steam generator tube denting problems which had been identified at other plants, Con Edison initiated, in 1977, an extensive program at Indian Point Unit 2 aimed at mitigating the effects of corrosion and minimizing corrosion rates. This program included state of the art steam generator tube examination techniques such as several different kinds of eddy current testing and the development of a technique called profilometry for measuring tube shapes

and calculating tube denting utilizing electro-mechanical sensors. (Rothstein Testimony, pp. 5-8) During the most recent refueling and maintenance outage which was completed in December, 1982, an extensive steam generator tube examination program was conducted which included the following examinations: (1) Eddy current examination of steam generator tubing for defects and dents; (2) Eddy current examination of selected tubes for inter-granular attack of the tube wall in the crevice; (3) Eddy current examination of selected tubes for pitting; (4) Profilometry examination of selected tubes to determine the strain resulting from tube distortion due to denting; (5) Visual, i.e., photographic, examination of the flow slots in the support plates of each steam generator; (6) Visual examination, as thoroughly as practicable, of the lower secondary side of all steam generators. (Id., pp. 8-10)

2.2.1-11 Based on steam generator tube examinations and operating experience at Unit 2, it appears that the corrosion of the steam generators at Indian Point 2, while progressive, is essentially arrested, that is, the rate of change between the two most recent examinations is hardly measurable. There is no pitting, cracking or other deterioration of any of the steam generator tubes. Tubes that have been significantly deformed as a result of accumulation of corrosion products between the tubes and sup-

port plates (tube denting) have been preventively plugged and removed from service. (Rothstein Testimony, pp. 12-13) At the present time there are 491 tubes plugged in the four steam generators at Unit 2. This constitutes less than four percent (4%) of the total number of steam generator tubes. Of the total number of tubes that have been plugged, approximately 400 had been plugged before the unit went into service, and less than 100 have been plugged during 10 years of operational service, preventively and to remedy a few tube leaks. (Id., p. 11; Rothstein, Tr. 6110) Based on the examination techniques utilized at Indian Point Unit 2 and the experiences at other facilities, it is improbable that any tubes with significant deformation have not been examined. Examination of the secondary side of the steam generators failed to reveal the presence of any significant foreign objects. (Rothstein Testimony, p. 13)

C. STAFF'S PROPOSED GENERIC REQUIREMENTS

2.2.1-12 The Staff's proposed program consists of the following:

(a) Steam Generator Integrity:

1. Prevention and detection of loose parts and foreign objects;

2. Stabilization and monitoring of degraded tubes;
3. Tube inservice inspection program (ISI);
4. Improved eddy current inspection techniques (ECT);
5. Primary to secondary leakage limit;
6. Secondary water chemistry program;
7. Condenser inservice inspection program;
8. Upper inspection ports;

(b) Plant Systems Response:

1. Reactor Coolant System (RCS) pressure control during a SGTR;
2. Safety injection (SI) signal reset;
3. Containment Isolation (CI) and reset;

(c) Radiological consquences:

1. Standard technical specification limit for coolant iodine activity.

(Rothstein Testimony, Attachment SR-2)

2.2.1-13 Each of the proposed draft SGOG steam generator generic requirements referred to in Board Question 2.2.1 was examined in the testimony and considered by the Board in light of the corrosion protection program now in existence at Indian Point Unit 2. In general, many of the proposed draft requirements are already being imple-

mented at Indian Point Unit 2. In fact, some of these items had their origin at Indian Point Unit 2. (Rothstein Testimony, p. 13) A point-by-point comparison of the current steam generator inspection program at Indian Point 2 with the SGOG generic draft requirements reveals that the Unit 2 program currently in place is generally equivalent to, and as a whole exceeds, the proposed draft generic requirements. (Id., pp. 13-27)

2.2.1-14 WBCA presented testimony which discussed the need for a secondary water chemistry program which was similar to that described in the proposed draft generic steam generator requirements. (Fleisher 2.2.1, pp. 1-2) After reviewing Con Edison's testimony with regard to the secondary water chemistry program currently in effect at Unit 2, WBCA's witness testified that he was not aware of any elements of the proposed draft requirements for secondary water chemistry programs which are not being accommodated by the current program at Indian Point Unit 2. (Fleisher, Tr. 6496) With specific regard to the secondary water chemistry program suggested in the proposed generic requirements, it is uncontroverted in the record that the current program requirements at Unit 2 exceed those in the proposed draft generic requirements. (Rothstein Testimony, p. 19; see Fleisher, Tr. 6497) The

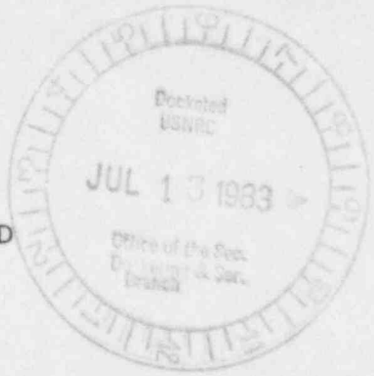
WBCA witness applauded the establishment of the secondary water treatment program at Unit 2 and the excellent record regarding lack of corrosion at that facility. (Fleisher, Tr. 6501)

2.2.1-15 Both the Staff and the Licensees agree that the requirements proposed at the July 29, 1982 meeting of the Staff and members of the SGOG should not be required for the Indian Point plants at this time. (Holahan/Rowsome Testimony, pp. 11-12; see Bley/Richardson 2.2.1, p. 6)

D. ULTIMATE FINDING

2.2.1-16 Based upon the testimony and cross-examination with regard to Board Question 2.2.1, the Board finds that the risk to the public health and safety associated with steam generator tube rupture events is extremely low. Further, the Board finds that the potential for significant steam generator tube degradation has been reduced by the aggressive corrosion protection program established by Con Edison at Indian Point Unit 2, and that the success of that program is evidenced by the current condition of the steam generators and their performance to date. The number of steam generator tubes which

have been plugged to date is very small, especially considering that most of the tubes which are now out of service were plugged during construction before the plant went into service. In addition, it should be noted that Indian Point Unit 2 remains one of the few pressurized water reactors that has not had a forced outage because of a tube leak. (Rothstein Testimony, p. 29) The Board finds that the proposed draft generic steam generator requirements discussed at the July 29, 1982 meeting with the Steam Generator Owners Group need not be imposed at Indian Point Unit 2 at this time because: (1) the imposition of such requirements would not perceptibly impact the risk to the public health and safety posed by the generator tube ruptures and (2) the program already in effect at Indian Point 2 is generally equivalent to and in many cases exceeds the proposed draft generic requirements.



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) : July 11, 1983

CON EDISON'S PROPOSED FINDINGS OF FACT REGARDING COMMISSION QUESTIONS 3 AND 4

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COMMISSION QUESTION 3

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FINDINGS OF FACT UNDER COMMISSION QUESTION 3

I. INTRODUCTION

3.0-1 Commission Question 3 reads as follows:

> What is the current status and degree of conformance with NRC/FEMA guidelines of state and local emergency planning within a 10 mile radius of the site and, of the extent that it is relevant to risks posed by the two plants, beyond a 10-mile radius? In this context, an effort should be made to establish what the minimum number of hours warning for an effective evacuation of a 10 mile quadrant at Indian Point would be. The FEMA position should be taken as a rebuttable presumption for this estimate.

3.0-2 The Board's February 7 Memorandum and Order Formulating Final Contentions on Commission Questions 3 and 4, admitted eight contentions for consideration in this proceeding. These contentions, i.e. Contentions 3.1, 3.2, 3.3, 3.4, 3.6, 3.7, 3.9 and 3.10 will be addressed seriatim. The Board will then make findings with regard to Question 3.

II. CONTENTION 3.1

3.1-1 Contention 3.1 reads as follows:

> Emergency Planning for Indian Point Units 2 and 3 is inadequate in that the present plans do not meet any of the sixteen mandatory standards of 10 CFR 50.47(b), nor do they meet the standards of Appendix E to 10 CFR Part 50.

3.1-2 Intervenors offered numerous witnesses who purported to address Contention 3.1. See "Index of Witnesses Presenting Testimony on Emergency Planning Issues on Behalf of Intervenors: UCS, NYPIRG, RCSE, WBCA, Parents, WESPAC, FOE, and Audubon and on Behalf of the Attorney General of the State of New York". The overwhelming majority of these individual pieces of testimony did not directly address the planning standards contained in 10 CFR §50.47(b) and in Appendix E to 10 CFR Part 50, nor did they address the question of whether the present plans for Indian Point meet these standards. In fact, many of these witnesses indicated a lack of familiarity with these standards, and in some instances, indicated that the intent of their testimony was not to allege compliance or non-compliance with these standards, but rather to indicate a belief that the emergency plans for Indian Point should be improved in some particular way. In addition, each of

the intervenor witnesses who is listed in the "Index of Witnesses . . ." as addressing Contention 3.1 addressed at least one and usually several other Contentions under Questions 3. In most cases this testimony more directly related to other more particular contentions under Question 3. However, intervenor testimony that is directly relevant to the question of whether the Indian Point emergency plans comply with the standards contained in 10 CFR § 50.47(b) or Appendix E to Part 50 has been considered in reaching these findings under Contention 3.1 even if the testimony did not directly allege non-compliance with a particular standard.

3.1-3 As we have noted previously (See November 15, 1982 Memorandum and Order, at 15), and as the interested states themselves have agreed, the emergency planning testimony they offered addressed for the most part Commission Questions 3 and 4, rather than the particular contentions admitted for litigation in this proceeding. Nevertheless, to the extent that witnesses of interested states offered testimony which is relevant to Contention 3.1, this testimony was considered in reaching the findings of fact for this contention.

3.1-4 Question 3 asks what is the "current status and degree of conformance with NRC/FEMA guidelines of state and local emergency planning." The great bulk of emergency planning testimony was submitted in June 1982. Some of the testimony offered under this contention has been mooted by the passage of time. That is to say, it is clear that in many instances concerns raised in testimony filed in June 1982 have since been addressed. The evidentiary record in this proceeding was closed some four months ago. The Board in these Findings of Fact is bound to consider only material that is part of the evidentiary record in this proceeding. As we have noted previously (November 15, Memorandum and Order at pp. 17-18) the Commission in weighing the Board's findings may consider such additional emergency planning information which it has received in the performance of its regulatory authority.

3.1-5 10 CFR § 50.47(b) lists sixteen emergency planning standards which both the "on-site and off-site emergency response plans for nuclear power plants must meet" in order for an applicant to obtain an operating license for a nuclear power reactor. These standards are made applicable to previously licensed, operating plants such as Indian Point pursuant to 10 CFR § 50.54(w). Of

these sixteen standards, fifteen provide requirements which are applicable to both the onsite emergency plans of licensees and the off-site state and local emergency plans. One standard, 10 CFR § 50.47(b)(2), relates solely to requirements regarding licensee plans.

3.1-6 Appendix E to 10 CFR Part 50 details the requirements that must be provided for in the emergency plan of an applicant for the construction permit for a nuclear power reactor (see Section IV, of Appendix E, "Content of Emergency Plans"). Thus the requirements of Appendix E relate to the provisions which must be provided for in a licensee's emergency plan and do not relate to the emergency plans of state and localities in the areas around nuclear power reactors.

3.1-7 These findings will deal separately with the Licensees' emergency plans and the off-site plans of New York State and the four counties around the Indian Point plants. The question of compliance of the state and county plans with the standards of 10 CFR § 50.47(b) will be considered first and the question of the compliance of Licensees' plans with these standards and with the standards of Appendix E to 10 CFR Part 50 will then be addressed.

A. STATE AND COUNTY EMERGENCY PLANS

3.1-8 Pursuant to Executive Order 12148, FEMA has been assigned lead responsibility for coordinating off-site radiological emergency response and preparedness planning. (FEMA Testimony, p. 1) The Nuclear Regulatory Commission's regulations, 10 CFR § 50.47(a)(2), provide that the NRC's findings in an operating license proceeding as to the adequacy of State and local emergency plans shall be based upon a review of FEMA's review of these plans. These regulations further provide that "[i]n any NRC licensing proceeding a FEMA finding will constitute a rebuttable presumption on a question of adequacy. . . ." (10 CFR § 50.47 (a)(2))

3.1-9 NRC and FEMA have developed guidance with regard to the 16 planning standards of 10 CFR § 50.47(b). This guidance is provided in a document entitled "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plant," NUREG-0654/FEMA REP-1, Rev. 1 (NUREG-0654). NUREG-0654 provides evaluation criteria for each of the sixteen planning standards. (FEMA Testimony, p. 3) The sixteen planning standards listed in 10 CFR § 50.47(b)(1)

through (16) have been designated by the first sixteen letters of the alphabet in NUREG-0654. Thus NUREG-0654 Planning Standard A is the same as 10 CFR § 50.47(b)(1) and Planning Standard P is the same as 10 CFR § 50.47(b)(16). There is a direct correspondence between a particular numbered clause of 10 CFR § 50.47(b) and the corresponding letter standard utilized in NUREG-0654. (Husar, Tr. 1339) Most of the testimony and exhibits introduced into this record nave utilized the NUREG-0654 planning standards. For purposes of these findings when reference is made to a particular planning standard a parenthetical reference will be made to the matching standard in 10 CFR § 50.47(b).

3.1-10 FEMA's proposed rule, 47 Fed. Reg. 48782 (1982) (to be codified at 44 CFR Part 50 (proposed October 28, 1982)) provides procedures whereby state and local plans for dealing with radiological emergencies will be reviewed by FEMA. Under the proposed rule, these plans are reviewed by a Regional Assistance Committee (RAC) which consists of representatives of various federal agencies. (FEMA Testimony, pp. 2-3)

3.1-11 On July 21, 1981 New York State submitted the generic State Radiological Emergency Response Plan to

FEMA for its review. On August 18, 1981 a State site-specific annex to the plan, which dealt specifically with the Indian Point plant, as well as annexes for Westchester, Rockland, Putnam and Orange Counties, were submitted to FEMA. (FEMA Testimony, p. 5) New York State Exhibit 3 represents the offsite radiological emergency response plans (RERPs) for Indian Point as they existed in August 1981. This nine-volume Exhibit includes the one-volume state plan (i.e., both the generic RERP as well as the site-specific portions of the state plan), as well as two-volume RERPs for each of the four counties. (Davidoff, Tr. 11295-11298) These plans were prepared under the direction of the New York State Disaster Preparedness Commission pursuant to that body's responsibilities under Article 2-B of the New York State Executive Law. The State's Radiological Emergency Preparedness Group (REPG) of which Mr. Davidoff is the Director, has a permanent staff which shoulders overall responsibility for administration of the plan. Mr. Davidoff testified that New York State Exhibit 3 was developed in accordance with the guidelines contained in NUREG-0654. (Davidoff/Czech Testimony, pp. 7-8)

3.1-12 On September 29, 1981 and December 31, 1981, FEMA transmitted comments on New York State Exhibit 3, including RAC comments, to New York State. (FEMA Testimony, p. 5) These comments are appended to the FEMA Initial Testimony as Attachment A and Attachment B respectively. New York State Exhibit 4, which was prepared by Mr. Davidoff, with the participation of Mr. Czech, consists of New York State's response to these FEMA comments. This response was sent by Mr. Davidoff to FEMA on June 25, 1982. (Davidoff, Tr. 11298-11301)

3.1-13 On March 3, 1982 an emergency planning exercise was conducted at Indian Point in order to test the onsite and offsite emergency plans and to determine the level of compliance with federal emergency planning standards. On May 27, 1982 FEMA issued an assessment of the offsite portions of the exercise. The document "Post Exercise Assessment, Exercise of the New York State and Westchester, Putnam, Rockland and Orange Counties Emergency Plans for Indian Point Nuclear Generating Station" (1982 Post Exercise Assessment) is Attachment C to the FEMA Testimony. (FEMA Testimony, p. 5) Based upon the 1982 Post Exercise Assessment FEMA concluded that at the time of the March 3, 1982 exercise there were significant planning

deficiencies in five of the fiteen planning standards applicable to offsite emergency planning.

3.1-14 New York State Exhibit 5, "Exhibit of New York State Indian Point 3 Exercise Schedule for Remedial Actions," was prepared under Mr. Davidoff's supervision. Exhibit 5 consists of New York State's item-by-item plan for remedial action in response to FEMA's comments on the March 1982 exercise. (Davidoff, Tr. 11300-11302)

3.1-15 On July 30, 1982, at the request of the NRC, FEMA issued "Interim Findings on the Adequacy of Radiological Emergency Response Preparation of State and Local Governments of the Indian Point Nuclear Power Station." (Interim Findings) The Interim Findings informed the NRC of the significant deficiencies in the five planning standards noted in the 1982 Post Exercise Assessment. As a result of the Interim Findings, on August 3, 1982 the NRC issued an order which notified Licensees that unless the significant deficiencies were corrected within 120 days, the NRC would consider whether enforcement action was appropriate. In response to this order a working group was established at the request of NRC and FEMA, which consisted of New York State, the Licensees and

representatives of the four counties (Rockland County's participation in this work was limited to being an observer). (Davidoff, Tr. 11323) This working group met regularly with NRC and FEMA in an attempt to resolve the deficiencies noted in the Interim Findings. The final report of this group, New York State Exhibit 7, is entitled "Interim Findings, Indian Point Radiological Emergency Preparedness Remedial Action Schedule." (Davidoff/Czech Supplemental Testimony, p. 2; Davidoff, Tr. 11305) This document presents the State of New York's response to each of the 34 elements of FEMA's five significant deficiencies.

3.1-16 On December 16, 1982 FEMA issued its "Update Report on the Status of Remedial Actions Cited in the July 30, 1982 Interim Findings on the Adequacy of Radiological Emergency Response Preparation of State and Local Governments at the Indian Point Nuclear Power Station" (December 16, 1982 Update Report). This document is attached to FEMA's Supplemental Testimony, ff. Tr. 14741. The December 16, 1982 Update Report concluded that all of the significant deficiencies noted in the Interim Findings had been adequately addressed with the exception of two areas. First, FEMA concluded that the absence of contractual arrangements with commercial bus drivers in

Westchester County constituted a significant deficiency in NUREG-0654 Planning Standard J "Protective Actions" (i.e., 10 CFR § 50.47(10)). (December 16, 1982 Update Report pp. 5-6) Second, FEMA concluded that due to Rockland County's lack of involvement in the emergency planning effort and the disavowal by that County of the emergency plans that had been prepared for it, a significant deficiency existed with regard to NUREG-0654 Planning Standard P "Responsibility for Planning Effort" (i.e., 10 CFR § 50.47(b)(16)). (Id., pp. 7-9)

3.i-17 On March 9, 1983 an exercise of the onsite emergency response plan for Indian Point 2 and the New York State and County emergency plans was conducted. On April, 14, 1983, FEMA issued a document entitled "Post Exercise Assessment of the Radiological Emergency Response Plans of New York State and Westchester, Rockland, Orange and Putnam Counties for the Indian Point Nuclear Power Station" (1983 Post Exercise Assessment). The 1983 Post Exercise Assessment, together with the April 14, 1983 letter of Frank Petrone, Regional Director for FEMA Region II, (Petrone Letter) transmitting it to FEMA's Washington headquarters, are attached to FEMA's Supplemental Testimony, ff. Tr. 14741. Both the Petrone Letter (p.1)

and the 1983 Post Exercise Assessment (pp. ix-xiii) noted a general, marked improvement in the level of preparedness in offsite emergency planning at Indian Point over that displayed in the March 1982 Indian Point 3 exercise. Nevertheless, based upon his review of the 1983 Post Exercise Assessment, Mr. Petrone determined that the two significant deficiencies noted in the December 16, 1982 Update Report, that is, the lack of contractual arrangements with Westchester County bus drivers and the question of Rockland County's involvement in emergency planning, remained. (Petrone Letter, pp. 1-3)

3.1-18 From the time of their submittal to FEMA in August 1981, the state and four county RERPs have undergone extensive revisions to reflect comments of FEMA and others as well as to incorporate changed circumstances. New York State Exhibit 6 constitutes revisions that were made to the State's Radiological Emergency Response Plan which have been submitted by the State to FEMA. (Davidoff, Tr. 11304-11305) A totally revised Putnam County Radiological Preparedness Plan, dated January 1983, is New York State Exhibit 11. (Tr. 11307-11308) Following its decision to reject the original Rockland County Radiological Emergency Response Plan, the

County set about to develop a revised plan. The New York State Radiological Preparedness Group provided assistance in this effort. (Davidoff, Tr. 11361-11362) The County has produced a draft plan which the State has adopted as a compensating measure to be used until the County completes the development of its own plan. (Davidoff, Tr. 11362, 11364) It was this plan which was exercised during the Indian Point 2 March 9, 1983 exercise. Mr. Davidoff testified that the adoption of the draft plan as a compensating measure was done with the consent of the County. (Tr. 11362-11363) New York State Exhibit 10 is the State's Compensating Measure for Rockland County Radiological Emergency Preparedness Plan. New York State Exhibit 12 consists of revisions, dated February 17, 1983, to the Orange County Radiological Emergency Response Plan which were prepared by Orange County staff with the assistance of the State's Radiological Emergency Preparedness Group. (Tr. 11311) New York State Exhibit 13 consists of revisions to the Westchester County Radiological Emergency Preparedness Plan dated February 1983. (Tr. 11312)

3.1-19 The FEMA testimony and the testimony of New York State REPG both assess compliance of the state and

county RERPs for Indian Point with the standards of 10 CFR § 50.47(b). (See Davidoff/Czech Testimony, p. 8; Davidoff, Tr. 11341) No other party to this proceeding attempted to conduct a detailed assessment of the New York State and four counties' emergency plans against these standards. Messrs. Schmer and Scalpi, the persons responsible for emergency planning in Orange and Putnam counties respectively, did, however, testify that the plans for their counties comply fully with NUREG-0654 standards. (Tr. 12108) Although witnesses for Westchester and Rockland with responsibilities in their counties similar to those of Messrs. Schmer and Scalpi did testify regarding the level of emergency preparedness in their respective counties, no attempt was made by these witnesses to measure the emergency plans for their counties against the standards of 10 CFR § 50.47(b).

3.1-20 Contention 3.1 alleges that emergency plans for Indian Point fail to meet any of the sixteen standards in 10 CFR § 50.47(b). In determining whether a particular standard is met this Board must make a reasoned, qualitative judgment. A standard can be met even though there are areas where improvements can and should be made. Similarly, as FEMA testified, it is possible for one or

more of the particular NUREG-0654 criteria elements which are used as guidelines in assessing compliance with a standard to be not satisfied and still find that the planning standard is satisfied. (Husar, Tr. 1338-1339) The evaluation criteria amplify and "expand on the standards", (FEMA Testimony, p. 3), set forth in 10 CFR § 50.47(b). A determination whether an emergency plan meets a particular overall planning standard requires an examination of the elements embodied within the particular planning standard. Many of the NUREG-0654 evaluation criteria address the details of emergency planning. For example, Evaluation Criterion (1)(C) under the NUREG-0654 Planning Standard A, i.e., (10 CFR § 50.47(b)(1)) calls for the inclusion of block diagrams in the emergency plans showing the division of responsibility among emergency response organizations. It is clearly possible that an emergency plan could meet Planning Standard A, which looks to the plan overall in order to determine whether the lines of responsibility among the various response organizations are established, even if one or all of the block diagrams called for by Criterion (1)(C) are not included in the plan.

10 CFR § 50.47(b)(1)

3.1.21 10 CFR § 50.47(b)(1) and NUREG-0654 Planning Standard A read as follows:

> Primary responsibilities for emergency response by the nuclear facility licensee and by State and local organizations within the Emergency Planning Zones have been assigned, the emergency responsibilities of the various supporting organizations have been specifically established, and each principal response organization has staff to respond and to augment its initial response on a continuous basis.

3.1-22 It will be seen that 10 CFR § 50.47(b)(1) requires that radiological emergency plans provide for a clear delineation of responsibilities between the various organizations with emergency response duties, including the provision of responsibilities for support groups which are to assist those organizations with primary responsibilities in particular areas. In addition, 10 CFR § 50.47 (b)(1) requires that the plans provide that each principal response organization have sufficient staff to respond to an emergency on a continuous basis.

3.1-23 The onsite plans of each licensee and the State and four County RERPs delineate the respective responsibility of licensees, state and local governments, (See, e.g., Con Edison Emergency Plan*, § 5.4; State RERP,

Part I, §§ I (D), II(B), III(B); Westchester County RERP, § I.E) The plans describe the functions and staffs of the various organizations. In addition, the plans identify the various support organizations which would be called upon in an emergency, including those of the federal government. (See, e.g., State RERP, p. III-12) In addition, the responsibilities of private organizations, including radiological laboratories, (see, e.g., Appendix A to Emergency Plans of Licensees), and the American Red Cross are identified. (See, e.g., State RERP, p. III-25, FEMA Testimony, p. 21)

3.1-24. The amendments to Article 2-B of the New York Executive Law, contained in Chapter 708 of the Laws of 1981, have clarified the responsibilities of state, county and local governments in planning for, and responding to radiological emergencies. Under the statute, although the

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* Con Edison's "Emergency Plan for Indian Point Unit Nos. 1 and 2" (Con Edison Emergency Plan) was admitted as an exhibit to Con Edison's onsite testimony. (Tr. 14480)

County executive has initial responsibility for the governmental response to an emergency, the Governor may declare an emergency and assume responsibility for coordinating the response of state, county and local governments. A copy of Article 2-B as amended is Power Authority Exhibit 41. (Admitted, Tr. 11443)

3.1-25 The capability of the State and the four counties to provide sufficient staff to respond to a radiological emergency on a continuous basis was demonstrated during the March 9, 1983 Indian Point exercise (1983 Post Exercise Assessment pp. 19, 33, 38, 46, 53) In addition, § 29 of the New York Executive Law permits the Governor to provide additional state personnel to respond to a radiological emergency.

3.1-26 In its December 16, 1982 Update Report FEMA reviewed the joint efforts of the state, counties and Licensees to respond to the comments previously submitted by FEMA with regard to NUREG-0654 Planning Standard A (i.e., 10 CFR § 50.47(b)(1)). The December 16, 1982 Update Report concluded, after reviewing the state and counties' plans, that there were no significant deficiencies in the state and county plans with regard to the planning

standard, although FEMA believed there were a number of deviations from the NUREG-0654 evaluation criteria in particular areas. The Update Report found that several of the previously noted deviations from the evaluation criteria had been corrected and that steps were underway to correct others. (See December 16, 1982 Update Report, pp. 12-13)

3.1-27 FEMA's assessment of the implementation of the state and county plans during the March 9, 1983 drill also found an overall high level of conformance with Planning Standard A. The Petrone Letter did not allege any significant deficiencies in this standard. The 1983 Post Exercise Assessment indicates some failures to fully meet applicable NUREG-0654 evaluation criteria in certain particulars. Each of these, however, relate to the overall issue of Rockland County's role in emergency planning. (See 1983 Post Exercise Assessment, Table 1) All other deviations from the NUREG-0654 evaluation criteria for the planning standard which were noted in the 1982 Post Exercise Assessment are shown by the 1983 Post Exercise Assessment to have been corrected. (Id.)

3.1-28 No probative evidence was offered which alleged non-compliance with the planning standard.

3.1-29 As noted earlier, (Finding 3.1-8), FEMA's findings with regard to the adequacy of emergency planning are entitled to presumptive weight in NRC licensing proceedings. Although this proceeding is not a licensing proceeding and thus FEMA's findings on this issue are not entitled to presumptive weight, the Board recognizes FEMA's expertise and will accord its findings significant weight when it is appropriate. Its findings with regard to overall compliance with 10 CFR § 50.47(b)(1) are in agreement with those of Mr. Davidoff, the New York State official with day-to-day responsibilities for radiological emergency planning, who has testified that the plans are in substantial compliance with all of the applicable standards. (Davidoff/Czech Testimony, p. 8; Davidoff, Tr. 11341) Similarly, officials responsible for emergency planning in Orange and Putnam counties concluded tnat the plans are in compliance with applicable standards. (Scalpi, Schmer, Tr. 12108) In light of this expert testimony and the Board's own review of the state and county plans, and in view of the absence of probative testimony alleging noncompliance with 10 CFR § 50.47(b)(1),

the Board finds that the state and county RERPs meet this standard.

10 CFR § 50.47(b)(2)

3.1-30 10 CFR § 50.47(b)(2) and NUREG-0654 Planning Standard B relate to licensee's emergency plans and are discussed at Findings 3.1-169 to 3.1-175.

10 CFR § 50.47(b)(3)

3.1-31 10 CFR § 50.47(b)(3) and NUREG-0654 Planning Standard C provide as follows:

> Arrangements for requesting and effectively using assistance resources have been made, arrangements to accommodate State and local staff at the licensee's near-site Emergency Operations Facility have been made, and other organizations capable of augmenting the planned response have been identified.

3.1-32 This planning standard deals with the provision in emergency plans of the means to obtain and accommodate emergency response support from state and local agencies as well as other organizations.

3.1-33 The State and County plans make explicit provision for calling upon the Federal response

capabilities embodied in the Federal Radiological and Monitoring Assistance Program, (see, e.g., State RERP, p. III-12 and Procedure H; Westchester RERP, pp. I-14 to I-15), provide for state and county representatives at Licensees' near-site emergency operating facility, (see e.g., State RERP, p. H-3; Westchester RERP, § III.D.2 (p.II-22)), and identify radiological laboratories and other organizations which can provide assistance. (See, e.g., State RERP, pp. III-5 to III-25, III-38 to III-39)

3.1-34 FEMA's December 16, 1982 Update Report found no significant deficiencies with regard to this planning standard, although it did note a number of areas where improvements could be made in areas related to particular NUREG-0654 evaluation criteria. (1982 Update Report, p. 14)

3.1-35 The 1983 Post Exercise Assessment did not note a single deficiency in this planning standard or in the evaluation criteria under the planning standard. The only deficiency referenced under this planning standard, a recommendation that the duties of the Rockland County representatives at the near-site EOF be more closely defined, was noted during the 1982 exercise. Compliance

with the recommendation was not assessed during the 1983 exercise since State personnel substituted for Rockland County personnel. The 1983 Post Exercise Assessment noted that the "overall facilities and resources were good at the EOF" (p. 21), and that the "active participation by all state, county and utility organizations at the EOF . . . was outstanding." (p. 22)

3.1.-36 The State described the provisions made for obtaining abundant federal assistance to provide additional monitoring capabilities through the Federal Radiological Monitoring Assistance Program, as well as from other sources. (Davidoff, Czech Tr. 11420-11422, 11437-11438)

3.1-37 No testimony was submitted by intervenors or interested states that contended 10 CFR § 50.47(b)(3) was not met. In light of the clear evidence to the contrary, the Board concludes that the standard is met.

10 CFR §50.47(b)(4)

3.1-38 10 CFR § 50.47(b)(4) and NUREG-0654 Planning Standard D provide as follows:

> A standard classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and state and local response plans call for reliance on information provided by facility licensees for determination of minimum initial off-site response measures.

3.1-39 To the extent that this standard relates to off-site emergency response plans, it requires that the plans provide for reliance on information provided by Licensees in determining initial emergency response.

3.1-40 Our review of the RERPs indicate that the plans do provide for reliance on Licensees' emergency classification and action levels. (See, e.g., State RERP, pp. III-27 to III-29, and Procedures B and D; Westchester RERP, § III-F (pp. III-31 to III-32) and Section 2 of Procedures 1 to 10) None of the FEMA reports in this record have found any deficiencies in the state and county plans with regard to this planning standard or with the evaluation criteria under it. In light of the above and the absence of any evidence to the contrary, the Board finds that the state and counties' RERPs are in compliance with 10 CFR § 50.47(b)(4).

10 CFR § 50.47(b)(5)

3.1-41 10 CFR § 50.47(b)(5) and NUREG-0654 Planning Standard E provide as follows:

> Procedures have been established for notification, by the licensee, of State and local response by all organizations and for notification of emergency personnel by all organizations; the content of initial and followup messages to response organizations and the public has been established and the means to provide early notification and clear instruction to the populace within the plume exposure pathway Emergency Planning Zone have been established.

3.1-42 This planning standard basically consists of three requirements: (a) the establishment of procedures for the notification of emergency personnel; (b) the provision of planned messages to be used in an emergency; and (c) the development of the means to provide early notification and instruction to the public within the plume EPZ.

3.1-43 The State and County RERPs provide detailed procedures for notifying emergency response personnel upon receipt of notification by the Licensee of the onset of any of the four emergency classification. (See, e.g., State RERP, Part I, § III (B)(3)(b) (pp. III-28 to III-33) and Procedures B and D; Putnam County RERP, §

III (f) (pp. III-21 to III-23) and Figure III-2 (p. III-41) Notification by the Licensees of the State and Counties would be effected through the Radiological Emergency Communication System (RECS), a direct line multiparty telephone system connecting licensees with governmental authorities. (Con Edison Onsite Testimony, pp. 12-13; Power Authority Onsite Testimony, p. 13; State RERP, Procedure B) Standard messages have been prepared for initial and follow-up messages from Licensees to offsite authorities. (Con Edison Onsite Testimony, p. 13) In addition, prepared forms are printed for messages to the public regarding possible protective action to be provided to the public. (See, State RERP, Procedure C, attachment 1; Westchester RERP, pp. PI/E 16-23, (as added by NYS Ex. 13))

3.1-44 Early notification of the public would be affected primarily through an Alert and Notification System (ANS) consisting of an extensive siren system around the Indian Point area. The siren system was installed to provide public alerting consistent with the guidance provided in NUREG-0654 (Con Edison Testimony p.13). Following the March 3, 1982 exercise the siren system was extensively enhanced. Existing sirens were upgraded, some

were relocated and additional sirens were installed. (FEMA Supplemental Testimony; Con Edison Supplemental Onsite Testimony, p. 1; Jackson, Tr. 11759) In addition, tone alert radios have been provided by Licensees to facilities such as schools and nursing homes. (See, e.g., Putnam County RERP, p. D-12)

3.1-45 The activation of the ANS provides notification to the public to listen to designated Emergency Broadcast System (EBS) radio and television stations which will give instructions as to what actions the public should take. (Davidoff, Tr. 11450-11451; UCS/NYPIRG Ex. 14, p. 4)

3.1-46 This standard was among the five planning standards which were identified as significantly deficient by FEMA in its July 30, 1982 Interim Findings. (See December 16, 1982 Update Report, p. 1) In the period following the issuance of the Interim Findings, significant efforts were made by the State, counties and Licensees in addressing each of the deficiencies noted by FEMA with regard to 10 CFR § 50.47(b)(5). A complete listing of the remedial actions undertaken to address each of the deficiencies in NUREG-0654 Planning Standard E (10 CFR §

50.54(b)(5)) is included in New York State Exhibit 7, pages 1 to 2. The December 16, 1982 Update Report found that all of the deficiencies which had been noted in the July 30 Interim Findings had "been adequately addressed." (December 16, 1982 Update Report, p. 4) The Update Report did, however, note areas in which FEMA believed the plans did not meet NUREG-0654 evaluation criteria in certain particular areas. The Update Report concluded that these deviations from NUREG-0654 constituted a minor deficiency. (Id., pp. 5-7)

3.1-47 The 1983 Post Exercise Assessment found marked improvement with this planning standard over the level displayed during the March 1982 Indian Point 3 exercise. The ability to alert and mobilize emergency response performance operations was rated either as acceptable, (1983 Post Exercise Assessment, pp. 25, 38), or good. (Id., pp. 46, 53) It was found that the sirens and tone-alert radios, which had been installed in schools, nursery school and special facilities, and the EBS system had all operated properly during the drill. (Id., pp. 26-27, 39-40, 47-48, and 53-54) In addition, all of the deviations from the NUREG-0654 evaluation criteria for this standard which had been noted in the 1982 Post Exercise

Assessment had been corrected with the exception of those that relate to Rockland County. (1983 Post Exercise Assessment, pp. 65-79)

3.1-48 Donald Smith, a witness for UCS/NYPIRG, offered testimony that is relevant to this planning standard, as well as to 10 CFR § 50.47(b)(7) which deals with public information and education. Mr. Smith, a professor of journalism and mass communications at the University of Iowa, concluded, based upon his review of NUREG-0654, portions of various Indian Point emergency planning documents and the pamphlet "Indian Point, Emergency Planning and You," that the means of informing and educating the public utilized at Indian Point are inadequate. (Smith Testimony, pp. 2-3)

3.1-49 Mr. Smith believes that in order to effectively communicate an emergency message to the public there must first be a detailed study of the intended audience followed by the formulation of communication strategies with different messages tailored for particular subgroups of the audience. (Id., p. 5) Mr. Smith testified that EBS messages should include provision for that portion of the population which may ignore an order to evacuate.

These messages would include instructions directed to these people telling them what to do if they choose not to evacuate. (Tr. 10344-10345)

3.1-50 Mr. Smith testified that he had no opinion as to whether the Indian Point information materials he examined comply with applicable NUREG-0654. (Tr. 10270) In addition he was unaware of emergency plans for any nuclear plant which reflect his view of what adequate public notification and information material should be. (Tr. 10279) Thus, his testimony suggests a belief that there is a need to revise existing guidelines related to notifying and informing the public rather than a conclusion that the emergency plans for Indian Point fail to meet NRC regulations on notifying and informing the public.

3.1-51 Although he believes a detailed assessment of the intended audience should be undertaken prior to developing a communication strategy, Mr. Smith was unaware that such an assessment had been undertaken by Licensees prior to developing the informational material he criticized. Roger Seasonwein, a witness for Assemblyman Brodsky, testified that a survey undertaken on behalf of Licensees to assist in developing communication strategies

to alert and inform the public was a "good survey." (Tr. 10694)

3.1-52 Mr. Smith stated that the EBS messages utilized in the March 1983 exercise were a "distinct improvement" over those used in the 1982 drill, and were in fact, "models of logic," but were still in his view "insufficient." (Tr. 10302)

3.1-53 Mr. Smith was unaware of any efforts on the part of New York State and Licensees to provide training to the media prior to the March 9, 1983 exercise. (Tr. 10333) Provisions are in fact made in the plans for annual briefings to press personnel. (See, e.g., Westchester RERP, pp. PE/I-3, PE/I-31) His testimony also indicates no familiarity with training provided to EBS station personnel. (NYS Ex. 7, p. 2)

3.1-54 The Board finds that Mr. Smith's testimony does not establish that the EBS system used at Indian Point fails to meet federal guidelines. In addition, the Board finds that it would approach the impossible to develop an EBS system which would meet Mr. Smith's standards and at the same time affect the desired goal -- the effective,

clear communication to the public of what it should do. Mr. Smith's suggestion that different EBS messages be transmitted for different audiences based upon psychological, sociological, gender and age differences, bespeaks a lack of familiarity with how the EBS system is designed to operate during a radiological emergency. As FEMA has noted, "EBS messages for evacuation and sheltering are necessarily quite lengthy due to the need to include descriptions of ERPA boundaries and evacuation directions." (1983 Post Exercise Assessment, p. 24) The development of ten or more EBS messages just for Westchester County portion of the EPZ was viewed by Mr. Smith as something that could be prepared for. (Tr. 10277-10278) The Board disagrees. Mr. Smith himself agreed that the use of separate messages could result in confusion. (Tr. 10362)

3.1-55 The Board also finds that it would be foolhardy to follow Mr. Smith's suggestion that the EBS messages include "presentation of options" to the public, including what to do if one chooses to ignore advice to evacuate. EBS messages should provide clear instructions to the public as to the preferred protective measures. Inclusion of options to these measures would imply approval of ignoring the suggested protective measures and would be counterproductive.

3.1-56 Based upon its review of the evidence in the record, including the portions of the emergency plans which relate to 10 CFR § 50.47(b)(5), and in the absence of material evidence to the contrary, the Board concludes that the plans meet the standard.

10 CFR § 50.47(b)(6)

3.1-57. 10 CFR § 50.47(b)(6) and NUREG-0654 Planning Standard F, read as follows:

> Provisions exist for prompt communications among principal response organizations to emergency personnel and to the public.

3.1-58 Extensive provisions have been made in order to insure prompt communication among the various response organizations that could be involved in responding to an emergency at Indian Point. In addition to the RECS system discussed in Finding 3.1-43, these facilities include: a) standard telephone lines; b) radio communication between onsite personnel and offsite governmental organizations; c) a hotline telephone network connecting Licensees' control rooms with the Emergency Operations Facility (EOF), the alternate EOF, the State Emergency Operations Center, the Counties Emergency Operations Centers and the City of Peekskill Emergency

Operations Center; and d) the National Warning System (NAWAS) Line connecting the control rooms and state and county warning points and the City of Peekskill (Con Edison Onsite Testimony, p. 14; Power Authority Onsite Testimony, pp. 28-29; Sears Testimony, pp. 18-20) The counties would also be able to utilize the Radio Amateur Civilian Emergency Services, an organization of "ham" radio operators who have been trained to respond to radiological and other emergencies. (FEMA Testimony p. 14; Marasco, Tr. 5498-5500; Schmer, Scalpi, Tr. 12097) In addition, the plans provide for the use of various other pre-existing communication systems within the different organizations (e.g. fire departments, civil defense organizations highway departments). (See, e.g., Putnam RERP, pp. C-9 to C-10; Orange RERP, p. E-1 to E-10; Westchester RERP, pp. E-9 to E-12; NYS Ex. 10 (i.e. New York State Compensating Measure for Rockland County), pp. III-3 to III-5)

3.1-59 FEMA's December 16, 1982 Update Report noted that there existed only a minor deficiency with regard to this planning standard. The Update Report showed that a number of FEMA's previous concerns with regard to this planning standard had been adequately addressed, although there remained a number of what FEMA considered to

be deviations from NUREG-0654 evaluation criteria in particular areas. (December 16, 1982 Update Report, pp. 19-20)

3.1-60 The 1983 Post Exercise Assessment showed marked improvement in this planning standard. Excepting those cases related to Rockland County, in each instance in which the May 27, 1982 Post Exercise Assessment found a deviation from a NUREG-0654 evaluation criteria, the 1983 Post Exercise Assessment showed the deviation to be corrected. (1983 Post Exercise Assessment, pp. 66-76) The comments in the Assessment regarding communications during the March 3, 1983 Indian Point exercise were generally laudatory and pointed to the improvements made since the 1982 exercise. (Id., pp. 18, 21, 25, 36-37, 46, 52-53)

3.1-62 A number of witnesses testified that not all of the local police departments within a particular county are connected via a Multi-District Radio (MDR) System and claimed that this could impede emergency planning at Indian Point. (See, e.g., Goldfarb Testimony, p. 1; Goldfarb Supplemental Testimony, p. 1; Marasco Testimony, p. 3) None of these witnesses established, however, that provision of this capability is required by

Commission regulations. In addition, there was no showing that the dedicated hotline now used to communicate among municipal police departments, (Marasco, Tr. 5463; Goldfarb Tr. 5775-5776), not connected by an MDR could not perform during a radiological emergency. In addition any suggestion that this or other suggested changes in communications capability is needed in order to assure effective communication during a radiological emergency is belied by FEMA's April 14, 1983 Post Exercise Assessment.

3.1-63 Based upon its review of the record the Board concludes that the State and County RERPs are in compliance with 10 CFR § 50.47(b)(6).

10 CFR § 50.47(b)(7)

3.1-64 10 CFR § 50.47(b)(7) and NUREG-0654 Planning Standard G read as follows:

> Information is made available to the public on a periodic basis on how they will be notified and what their initial actions should be in an emergency (e.g. listening to a local broadcast station and remaining indoors), the principal points of contact with news media for dissemination of information during an emergency (including the physical location or locations) are established in advance, and procedures for coordinated dissemination of information to the public are established.

3.1-65 This planning standard basically imposes three requirements with regard to providing the public information related to a radiological emergency: (a) provision of information on a periodic basis explaining how the public would be notified in case of an emergency and what actions the public would be expected to take; (b) establishment of points for dissemination of information to the news media in case of an emergency; and (c) development of procedures for the dissemination of information to the public.

3.1-66 The public information pamphlet "Indian Point, Emergency Planning and You" has been developed for annual distribution within the Plume Exposure Pathway EPA. The 1982 version of the pamphlet, (FOE Ex. 1), was distributed in early 1982. FEMA concluded that the pamphlet contained all required information. (FEMA Testimony, p. 17) As part of the overall effort after the March 1982 exercise to improve the Indian Point public information program the pamphlet was revised. This revision was a joint effort of Licensees, the State and the public information officials of Westchester, Putnam and Orange Counties. (Davidoff, Tr. 1331) The revision of the Rockland County pamphlet must await the revision of the County's plan (Davidoff/Czech Supplemental Testimony,

p. 11; Petrone Letter, p. 2) The revised brochures were distributed in Orange and Putnam Counties prior to the March 9, 1983 exercise. (1983 Post Exercise Assessment, pp. 48, 54) Distribution of a revised Westchester pamphlet was delayed, with FEMA's concurrence, until after the March 1983 exercise in order to reflect recent changes to the Westchester Plan. (Petrone Letter, p. 2; O'Rourke Testimony, p. 4) FEMA, (Kowieski, Tr. 15008), and Intervenor witness Smith, (Tr. 10335-10338), noted improvements in the 1983 version of the pamphlet.

3.1-67 Both the 1982 and 1983 versions of the pamphlet inform the public how they will be notified in case of an emergency and what they would be expected to do during an emergency. The pamphlets tell the public to turn to one of the listed EBS stations upon the sounding of the radiological emergency sirens. (FOE Ex. 1, p. 4; UCS/NYPIRG Ex. 14, p. 4) Instructions are provided regarding sheltering (FOE Ex. 1, p. 5; UCS/NYPIRG Ex. 14, p. 5) and evacuation. (FOE Ex. 1, pp. 6-14; UCS/NYPIRG Ex. 14, pp. 6-14) The evacuation portions of the pamphlet provide information regarding individual Emergency Response Planning Areas, specific information for the transit-dependent, bus pick-up points, designated

evacuation routes, and information related to reception and school reception centers, as well as details related to the evacuation of schools, nursing homes, hospitals and the disabled. In addition, the pamphlets provide instruction on preparation for an emergency, (i.e. "What you should do now," FOE Ex. 1, p. 15; UCS/NYPIRG Ex. p. 15), and background information on radiation and on Indian Point. (FOE Ex. 1, pp. 16-18, 20; UCS/NYPRIG Ex. 14, pp. 16-18, 20) The pamphlets contain a postage-paid postcard for persons such as the disabled or hearing impaired who believe they require special attention. In addition, toll-free numbers and addresses where additional information may be obtained are listed.

3.1-68 In addition to the public information pamphlet the current Indian Point public information program includes other methods to reach the public. These include the posting of signs and posters to inform transients, the placement of advertisements in local newspapers and on local radio stations, the development of inserts to be placed in telephone directories later this year, a speaker program and meetings with community groups, visits by the public to the plants, and a program to reach non-English speaking persons. (NYS Ex. 7, pp. 3-4; 1983

Post Exercise Assessment, pp. 4-5; Smith, Tr. 10294-10295; Jackson, Tr. 11743-11746; Schmer, Tr. 12130, 12186; Scalpi, Tr. 12130, 12186; Davidoff Tr. 11484; FEMA Supplemental Testimony, p. 5)

3.1-69 The RERPs provide for the operation of a joint news center in Verplanck, New York and for the designation of public Information Officers as the sources for official information during an emergency. (See, e.g., Westchester RERP, § III D.2-(c); Putnam County RERP § III, D.2(c); State RERP, p.III-15) In addition, the RERPs have been revised to provide detailed public information procedures. (See, e.g., State RERP, Procedure C, "Public Information,"; Westchester RERP, Procedure 16, "Public Information and Education")

3.1-70 The December 16, 1982 Update Report concluded that deficiencies in the Indian Point public information program which had been noted in the Interim Findings had been adequately addressed. The Update Report noted the comprehensiveness of this program and its use of several different methods to reach the public. (Update Report, pp 4-5)

3.1-71 The 1983 Post Exercise Assessment found that the Public Information Officer at the State Emergency Operation Center during the March 3, 1983 Indian Point 2 exercise was well-informed. (1983 Post Exercise Assessment, p. 20) The Assessment concluded, with regard to the Joint News Media, that "[o]verall, the emergency public information system functioned in a timely manner" and that "the system for handling media relations through the Joint Media Center . . . was very effective." (Id., p. 24) It noted that the rumor control system had operated well during the exercise. (Id., pp. 27, 41)

3.1-72 "Public and media relations" were found by the 1983 Post Exercise Assessment to be acceptable in Orange County, (Id., p. 48), and good in Putnam County, (Id., 54), during the March 1983 exercise. Posters and revised emergency planning brochures had been distributed in both counties. (Id., pp. 48, 54)

3.1-73 The 1983 Post Exercise Assessment concluded that public education in Westchester and Rockland Counties needed improvement. (Id. pp. 27, 40-41) The Assessment noted that a revised public information pamphlet had not been distributed at the time of the exercise in these counties and that posters apparently had not yet been

distributed. (Id.) The Assessment also stated that a spot-check on the day of the exercise indicated a lack of awareness on the part of the public as to what they should do in the event of an emergency. (Id.)

3.1-74 The Petrone Letter notes that public information posters had been completed and distributed in Westchester subsequent to the preparation of the Post Exercise Assessment. (Petrone Letter , p. 2)

3.1-75 FEMA testified that basically two factors were utilized in deciding whether the 1983 Post Exercise Assessment would recommend improvements in a county's public education program: (a) whether a majority of the respondents to FEMA's spot checks conducted on the day of the exercise knew what the correct response to a siren was; and (b) whether or not a revised brochure had been issued. (McIntire, Tr. 14953-14954) Mr. McIntire of FEMA agreed, however, that the spot checks were not statistically valid surveys. (Tr. 14952) He further testified that NUREG-0654 does not require that a majority of the persons surveyed on the day of an exercise know what to do in response to a siren, (Tr. 14954), and stated that he is unaware of how other FEMA regions attempt to assess public awareness. (Tr. 14954-14955)

3.1-76 Donald Smith, a UCS/NYPIRG witness, suggested that separate emergency planning pamphlets be prepared for different sub-groups within the population. (Smith Testimony, p. 12; Tr. 10338) As in the case of his suggestion regarding different messages to different population groups, (see Finding 3.1-50), the Board finds this suggestion to embody not so much an allegation that the pamphlets fail to meet Commission regulations as a suggestion that these regulations be revised. Also, as in the case of his suggestion that separate EBS messages be developed, (see Finding 3.1-54), we find that his suggestion is impractical. Finally, Mr. Smith's criticism that the Indian Point public information program over-relies on a single device, i.e. the pamphlet, (Smith Testimony, p. 7, Tr. 10294), is contradicted by the evidence that several methods are used to inform and instruct the public.

3.1-77 The Seasonwein Testimony, offered by Assemblyman Brodsky, and the Altschuler Testimony, offered by UCS/NYPIRG, both presented the results of surveys of Westchester residents to determine their familiarity with and attitude to emergency planning at Indian Point. We find that this testimony has no relevance to the question

of compliance with 10 CFR § 50.47(b)(7). Although a review of the record indicates a number of faults in the surveys of these witnesses, even if their respective surveys are correct there is no evidence to suggest that the survey results are indicative of a failure to comply with 10 CFR § 50.47(b)(7). There is no regulatory requirement that it be shown that all of the persons within the plume exposure pathway EPZ or even that a particular percentage of these persons show a particular level of understanding of emergency planning. (Sears Testimony, p. 58; McIntire, Tr. 14954-14955)

3.1-78 The Board concludes that the state and counties' plans for public information and education meet the requirements of 10 CFR § 50.47(b)(7). There has been a marked and steady improvement in performance in this area. FEMA does not currently believe there is a significant deficiency in this area. FEMA's concern in its 1983 Post Exercise Assessment with the extent of public understanding in Westchester and Rockland Counties is based upon "spot check" surveys which FEMA does not believe to be statistically valid and the fact that a revised pamphlet had not been distributed at the time of the 1983 exercise in these counties. Finding 3.1-66 deals with the reasons

why the revised pamphlet was not distributed in Westchester and Rockland Counties prior to the 1983 exercise.

10 CFR § 50.47(b)(8)

3.1-79 10 CFR § 50.47(b)(8) and NUREG-0654 Planning Standard H read as follows:

> Adequate emergency facilities and equipment to support the emergency response are provided and maintained.

3.1-80 Licensees have established a near-site Emergency Operations Facility at the Indian Point Service Center, in Buchanan, New York, approximately one-half mile from the site of the plant. An alternate site is located at Con Edison's Eastview Service Center, outside the EPZ. The EOF is the focal point for the collection, analysis and dissemination of radiological and meteorological information. The facility would house the Licensees' Emergency Director, as well as County, State and Federal representatives and would thus be the facility from which overall emergency response would be coordinated. (Power Authority Onsite Testimony, p. 18; Con Edison Onsite Testimony, pp. 15-16; Con Edison Emergency Plan, pp. 7-1 to 7-2)

3.1-81 Each County in the EPZ has an Emergency Operations Center (EOC). (See, e.g., Westchester RERP, § III D.2.a) In addition, provision is made for a State EOC in Albany, as well as an EOC at the Southern District Office of State Office of Disaster Preparedness in Poughkeepsie, to coordinate state activities. (State RERP, Part III, § I.D, pp. D-1 to D-9)

3.1-82 The 1983 Post Exercise Assessment found that each of the deviations from NUREG-0654 evaluation criteria applicable to this planning standard, which had been noted in the 1982 Post Exercise Assessment, had been corrected. (1983 Post Exercise Assessment, pp. 65-78) The 1983 Post Exercise Assessment found no new deviations from these criteria. In addition, the assessment found the facilities and resources at the Southern District and State EOCs, (p. 18), the near-site EOF, (p. 21), and in Westchester, (p. 25), Orange, (p. 45), and Putnam, (p. 52), to be good, while those of Rockland were rated as acceptable. (p. 36)

3.1-83 Licensees have provided much of emergency preparedness equipment utilized by the State and counties. Equipment provided by Licensees to the State and four

counties includes telephones and other dedicated communications systems; an augmented national alert warning system (NAWAS); single station "tone alert" radios; the RECS line; radiation monitoring kits; fixed offsite radiation monitoring systems; paging devices; a 146-unit siren system; and MIDAS computer terminals to allow authorities direct and independent access to critical data. (Power Authority Onsite Testimony, p. 26; J. McGuire, Tr. 3795; D. McGuire, Tr. 4125-4126, 4169; Curran, Tr. 5075, 5085; Davidoff, Tr. 11454; Schmer, Tr. 12094; Scalpi, Tr.12095)

3.1-84 The counties have received funds from the state to assist them in acquiring equipment which could be utilized in a radiological emergency. Much of these state funds were received by the state pursuant to the $250,000 annual fee assessed against Licensees and other reactor operators by § 29-c of the Executive Law. (Davidoff, Tr. 11376-11377, 11452) In addition, the state used portions of the licensee fees to directly provide equipment and training to the counties. The State has, for example, provided dosimeters, thermoluminescent dosimeters and potassium iodide tablets to the counties. (Davidoff/Czech Supplemental Testimony, pp. 3-4; Davidoff, Tr. 11391-11392)

3.1-85 In addition, Licensees have paid $210,000 to the State in 1982 pursuant to a contract with the State. All of the money paid pursuant to this contract was passed on to the four counties to further upgrade the status of emergency planning. (Davidoff/Czech Testimony, p. 3) This contribution was in addition to the $500,000 already paid that year pursuant to § 29-c of the Executive Law to the State for emergency planning that year, and was not required by law. (Davidoff, Tr. 11390-11391, 11453) In general, Licensees have been receptive to requests for radiological emergency planning equipment which they have received from the counties. (Scalpi, Tr. 12095; Schmer, Tr. 12093-12094)

3.1-86 Based upon our review of relevant evidence in the record the Board concludes that the emergency plans of the state and four counties meet 10 CFR § 50.47(b)(8).

10 CFR § 50.47(b)(9)

3.1-87 10 CFR § 50.47(b)(9) and NUREG-0654 Planning Standard I read as follows;

> Adequate methods, systems and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use.

3.1-88 Primary responsibility for the initial assessment of the magnitude and consequences of radiological accidents at Indian Point including offsite monitoring rests with Licensees. However, the plans contemplate the coordinated use of Licensee, state, county and federal resources, including the resources of the Radiological Assistance Plan and the Interagency Radiological Assistance Plan, administered by the Brookhaven area office of the United States Department of Energy. The various agencies relied upon and the different types of resources available are fully described in the RERPs. (See, e.g., State RERP, pp. III-34 to III-43, Procedure H; Westchester County RERP, pp. III-16 to III-17, III-38 to III-45, Appendix J)

3.1-89 Licensees' emergency plans have made provision for offsite monitoring. Section 7.3.2 of the Con Edison Emergency Plan and Section 7.4.2 of the Power Authority Plan describe facilities and equipment for offsite monitoring. Licensees have also installed a Reuter Stokes Sentri 1011 Remote Environmental Radiation Monitoring System consisting of 16 monitors within a 2-mile radius of the plants with the capability to telemeter radiation readings to a processor which relays it to the

Meteorological Information and Dose Acquisition System (MIDAS) computer system at the EOF. The MIDAS data can be accessed at County EOCs as well as data from the Atmospheric Release Advisory Capability (ARAC) system. The MIDAS and ARAC systems permit the rapid prediction of plume movement and dose assessment. In addition, Ludlum area radiation monitors are located at approximately 25 locations which are manned on a 24-hour basis. The Con Edison Nuclear Environmental Monitoring Group conducts a continuous monitoring program for the site and the surrounding area. (Con Edison Emergency Plan, § 7.3.2; Con Edison Implementation Procedure IP-1015; Power Authority Emergency Plan, § 7.4.2; Power Authority Procedure IP-1011; Power Authority Onsite Testimony, pp. 22-23; Con Edison Onsite Testimony, pp. 16-17; Sears Testimony, pp. 25-26; Curran, Tr. 5085; Schmer, Scalpi, Tr. 12204-12205)

3.1-90 In furtherance of their offsite monitoring role, Licensees have maintained an extensive system for the study and analysis of the meteorology of the Indian Point area. The meteorology of the area has been extensively studied both by consultants under contract to Licensees as well as Licensees' meteorologists. Two towers, the primary 122-meter tower and a ten-meter back-up tower, are located

onsite. Both of these towers have been instrumented in accordance with NRC Reg. Guide 1.23, "Meteorological Program in Support of Nuclear Power Plan," and NUREG-0654. (Con Edison Onsite Testimony, pp. 20-21) The 122-foot tower provides real-time meteorological data. Readouts of wind speed and direction and the ability to calculate Pasquill Stability are available in the Control Room. Access to this data is also available remotely by telephone. This data is sent to the MIDAS and ARAC Systems in the EOF. The ten-meter tower provides real time wind speed and direction data which are continuously recorded and sent to MIDAS. This data is available in the EOF and the EOCs and can also be accessed remotely by telephone. (Con Edison Procedure IP-1016; Power Authority Procedure IP-1003; Power Authority Onsite Testimony, pp. 23-24; Con Edison Onsite Testimony, p. 16; Sears Testimony, pp. 26, 51)

3.1-91. Further monitoring assistance would be available during an emergency. Con Edison provides for over 50 dose assessment and field monitoring teams. (Westchester RERP, p. J-8; Curran, Tr. 5076-5077; Jackson, Tr. 11760) In addition, the RERPs provide for obtaining extensive federal monitoring assistance. (See, e.g., State

RERP, pp. II-2 to II-3, Appendix H; Westchester RERP, pp. J-10 to J-22; Curran Tr. 5076; Davidoff, Tr. 11437-11438) The four counties also have independent field monitoring capability. (Scalpi, Schmer, Tr. 12151-12152; Curran, Tr. 5076; J. McGuire, Tr. 3794) In addition, the state can also provide radiological monitoring capability. (State RERP, p. H-4)

3.1-92 FEMA's 1983 Post Exercise Assessment indicated marked improvement in the level of performance related to this planning standard over that exhibited in the March 1982 exercise at Indian Point. In all instances in which there was a FEMA observance of an area under the Planning Standard in which the 1982 Post Exercise Assessment had indicated a failure to fully meet NUREG-0654 evaluation criteria, Table I of the 1983 Post Exercise Assessment shows that the deviation has been corrected. (pp. 65-79) A single deviation, relating to the need for additional training of Rockland County personnel, (1983 Post Exercise Assessment, p. 75), is shown. The 1983 Post Exercise Assessment found that accident assessment functions conducted at the Emergency Operations Facility and in Westchester, Rockland and Orange Counties were good. (Id., pp. 23, 28, 41, 49) Putnam County was rated as acceptable. (Id., p. 55)

3.1-93 The Board finds, based upon its review of the relevant evidence in the record, that the state and county plans are in compliance with 10 CFR § 50.47(b)(9).

10 CFR § 50.47(b)(10)

3.1-94 10 CFR § 50.54(b)(10) and NUREG-0654 Planning Standard J read as follows:

> A range of protective actions have been developed for the plume exposure pathway EPZ for emergency workers and the public. Guidelines for the choice of protective action during an emergency, consistent with Federal guidelines, are developed and in place, and protective actions for the ingestion exposure pathway EPZ appropriate to the locale have been developed.

3.1-95 The State RERP and each of the four County RERPs provide for a range of protective actions to be taken within the plume exposure pathway EPZ and detailed guidance to be used in determining which action is appropriate under particular circumstances. In addition, protective action options and guidance are provided regarding the ingestion pathway EPZ. In reaching a decision as to which protective action to implement, the RERPs adopt and incorporate the Protective Action Guides of the Environmental Protection Agency. (See, e.g., State RERP, pp. III-40 to III-41; Westchester RERP, p. III-40) Plant conditions,

meteorological data, projected doses, and time frames for the implementation of different protective action and for the termination of the incident would all be considered in selecting appropriate options. (See, e.g., State RERP, pp. III-39 to III-43; Westchester RERP, pp. III-39 to III-43)

3.1-96 The protective action options in the RERPs are complementary and additive. More than one protective action could be implemented at one time and different options could be selected for different populations (See, e.g., State RERP, p. III-43; Westchester RERP, p. III-46) Protective action options within the plume exposure pathway EPZ are: 1) Initial Precautionary Operations; 2) Selective Sheltering; 3) General Sheltering; 4) Selective Evacuation; and 5) General Evacuation. (State RERP, p. III-49)

3.1-97 The Initial Precautionary Operations response is an initial, relatively easy to implement response which would facilitate the implementation of other options if they become necessary. This option could include the temporary closing of schools and tourist areas, the establishment of traffic check points into the plume exposure EPZ and the activation of State and County EOCs. The option would decrease the number of persons within the

EPZ whose needs would have to be addressed if other protective actions become necessary while increasing the number of emergency response personnel. (See, e.g., Westchester RERP, pp. II-47 to III-48)

3.1-98 Selective sheltering would involve sheltering of those individuals who, because of physical or psychological conditions, would have difficulty dealing with evacuation. (Id., p. III-49)

3.1-99 General sheltering would involve a decision that all of the public within the EPZ or all of the public within certain affected areas be directed to take shelter. General sheltering would be selected when there is no need or insufficient time to evacuate. It could, for example, be utilized in the event of a puff-type release. (Id., pp. III-50 to III-51) It could also well be the preferred option in other circumstances, (Curran, Tr. 5157-5158), such as in a fast-moving accident situation. (Keller, Tr. 14856; see also Findings under Contention 4.2(b))

3.1-100 Selective evacuation would involve the evacuation of persons within all or part of the EPZ who

have a low tolerance for radiation. (See, e.g., Westchester RERP, pp. III-52 to III-53)

3.1-101 General evacuation would involve evacuation of the general public within all or parts of the EPZ (Id., pp. III-54 to III-56, Appendix A)

3.1-102 The provisions in the RERPs related to evacuation and the studies underlying these provisions are discussed in Findings under Contention 3.3, 3.6 and 3.9.

3.1-103 Provision is made for use of potassium iodide for emergency workers and staff and patients or inmates of facilities where evacuation is not possible or feasible. State RERP, pp. III-55 to III-55a; Davidoff/Czech Testimony, p. 16; Davidoff/Czech Supplemental Testimony, p. 12; see also Findings under Contention 4.2(a))

3.1-104 The State RERP identifies protective measures to be used for the ingestion pathway EPZ. These measures include environmental surveillance, actions to protect milk, dairy and other agricultural products, protection of water sources and coordination with

government agencies outside of New York State. (State RERP, pp. III-40, III-43 to III-49) Large numbers of trained personnel would be available to collect water and milk samples. (Czech, Tr. 11420-11421) The County RERPs also provide for implementing ingestion pathway EPZ protective actions. (See, e.g., Westchester RERP, pp. III-57 to III-58; Orange RERP, pp. III-56 to III-57)

3.1-105 In its December 16, 1982 Update Report FEMA found that all but one of the deficiencies in this planning standard which had caused compliance to be rated significantly deficient in the Interim Finding had been adequately addressed. The remaining deficiency related to the possible non-response of Westchester commercial bus drivers. It was because of this issue that the Update Report rated compliance with the Planning Standard to be significantly deficient. (December 16, 1982 Update Report, pp. 5-7)

3.1-106 The Petrone Letter concluded that the significant deficiency in this Plannning Standard "earlier identified" i.e., the possible non-response of bus drivers in Westchester, remains "significantly deficient." (Petrone Letter, p. 2; McIntire, Tr. 14772-14773)

3.1-107 The determination that the possible non-participation of Westchester commercial bus drivers was a significant deficiency was made by Mr. Petrone. (McIntire, Tr. 14781, 14962) The FEMA team leader in Westchester County who evaluated the March 9, 1983 exercise found that the ability to implement "Actions to Protect the Public" was "acceptable." (Kowieski, Tr. 14773-14774) He also found that, given the requirements of the exercise scenario, evacuation and transportation actions were adequate. (McIntire, Tr. 14893) FEMA witness McIntire agreed with this conclusion. (Id.)

3.1-108 FEMA testified that concerns expressed by County Executive O'Rourke about the bus issue in Westchester County also entered into Mr. Petrone's decision. (McIntire, Tr. 14798-14799) These concerns were expressed by the County Executive prior to the adoption of the "O'Rourke Plan" under which it is planned that children would be dismissed at the alert stage of a radiological emergency. (McIntire, Tr. 14800-14801) Mr. O'Rourke testified that the early dismissal plan solves some of the County's most fundamental concerns related to emergency planning. (O'Rourke Testimony, p. 3)

3.1-109 In considering FEMA's concerns about the possible non-response of bus drivers the Board notes that the President of Chappaqua Bus Company, one of the bus lines designated in the Westchester RERP, testified that he believed that most of his drivers would perform their duties in an emergency. (Corwin Testimony, p. 1) In addition, a bus driver testified that five of six drivers asked in her garage stated that they would respond to an emergency. (Narod-Shiek, Tr. 10283) In Putnam County 88 of 100 bus drivers sampled indicated they would respond to an emergency. (Scalpi, Tr. 12135)

3.1-110 In reaching its conclusion that a significant deficiency existed in Westchester County due to the possible non-response of commercial bus drivers FEMA did not consider the possibility that car pooling, ride-sharing and other measures might substantially reduce the number of persons who would be transit-dependent. FEMA agreed, however, that if there were such a reduction, this would be a major factor in determining the significance of this deficiency. (McIntire, Tr. 14809) Parsons Brinckerhoff, Quade & Douglas, Inc., the firm that drafted the evacuation plan portion of the RERPs, conservatively overestimated the transit-dependent portion of the

population by not accounting for carpooling and ridesharing. (Parsons Supplemental Testimony, pp. 13-14) A June 1980 survey reported that 74% of the prospective Indian Point evacuees with cars would be willing to pick up those without automobiles. (Id., p. 13) Mr. Davidoff testified that he believed the percentage of the population assumed by the RERPs to be transit dependent is too high for the Indian Point area. (Davidoff, Tr. 11471) Both the 1982 and 1983 versions of the Indian Point emergency planning pamphlet encourage ride-sharing. (FOE Ex. 1, pp. 6, 8; UCS/NYPIRG Ex. 14, pp. 6, 8) A survey introduced by intervenors indicates that only 3% of the population within the Westchester portion of the EPZ believes they would require bus transportation in the event of an emergency. (Seasonwein Testimony, p. 1-17)

3.1-111 There appears to be no uniform FEMA policy regarding the extent to which assurances of the availability of transportation services for the transit-dependent will be required for plan approval. FEMA Region 9 has found the transportation component of an emergency plan to be acceptable despite the absence of agreements with bus companies. (Con Edison Ex. 20, p. 10; see also FEMA Answer to Licensees Interrogatory 6, pp.

2-6) FEMA Region II did not receive any guidance from FEMA's national office on how to deal with the lack of written agreements. (McIntire, Tr. 14903)

3.1-112 FEMA rejected that possibility of using National Guardsman as substitutes for Westchester commercial bus drivers who failed to respond to a radiological emergency because in a worst case situation, i.e., an evacuation of all parts of the 10 mile EPZ and no response by any commercial bus drivers in Westchester County, the 4 to 5 hours needed to have the Guard in place was considered to be inadequate. (McIntire, Tr. 14812) FEMA has told the Commission that concerns about a "fast moving accident" led to the rejection of the National Guard option even though the additional time needed to mobilize the Guard would not be a concern in "other scenarios." (Con Edison Ex. 21, p. 12)

3.1-113 In making its finding with regard to the significance of the lack of commitments from bus operators FEMA did not attempt to determine significance in terms of public health, (McIntire, Tr. 14831-14832), and did not consult with the NRC in making this finding. (McIntire, Tr. 14830-14831) No attempt was made to assess the impact

of this deficiency on evacuation time estimates. (McIntire, Tr. 14865)

3.1-114 FEMA agreed that in the face of a fast-moving type of accident it would be possible that sheltering and not evacuation would be the preferred protective action (Keller, Tr. 14856), and that in a sheltering followed by relocation scenario the urgency for bus transportation would be substantially reduced from that in an evacuation before release scenario. (Keller, Tr. 14859)

3.1-115 The Board finds that the New York State REPG finding that this planning standard is met at Indian Point, (Davidoff/Czech Testimony, p. 8; Davidoff, Tr. 11341), is supported by the record. There is no basis for surmise that bus driver availability is any less certain in Westchester County than at other sites. By not accounting for ride-sharing, the RERPs have overstated, perhaps significantly, the demand for public transportation services in the event of a radiological emergency. Thus, FEMA's finding that the possible non-response of commercial bus drivers in Westchester County is a significant deficiency is unsupported in the record. In addition, we

also note our Finding 1.1-268, in which we found that in a fast-moving accident of the sort that led FEMA to reject the use of the National Guard in a "worst-case" situation, sheltering would provide protection equivalent to that afforded by evacuation.

3.1-116 Based upon its review of evidence in the record, the Board finds that the State and County RERPs meet 10 CFR § 50.47(b)(10).

10 CFR § 50.47(b)(11)

3.1-117 10 CFR § 50.47(b)(11) and NUREG-0654 Planning Standard K read as follows:

> Means for controlling radiological exposures in an emergency, are established for emergency workers. The means for controlling radiological exposures shall include exposure guidelines consistent with EPA Emergency Worker and Lifesaving Activity Protective Action Guides.

3.1-118 Both the State RERP and the County RERPs have detailed procedures for controlling and minimizing radiological exposure of emergency workers which are consistent with EPA and other federal guidelines. (See, e.g., State RERP, pp. III-56 to III-59 and Procedure G; Orange County RERP, pp. III-26 to II-29)

3.1-119 New York State Exhibit 8 shows that the State had distributed 950 self-reading dosimeters as well as 5500 permanent record dosimeters (TLD's) to the four counties as of the time of the preparation of the state's supplemental testimony. (Davidoff, Tr. 11322-11323)

3.1-120 In addition to the dosimeters that are shown in New York State Exhibit 8, there are many other dosimeters which are available in the EPZ, including hundreds of civil defense dosimeters, which are kept at the emergency preparedness offices in each of the four counties for distribution to emergency workers. In addition, there are thousands of these types of dosimeters stored at the regional offices of the Office of Disaster Preparedness, including the District Southern District Office of Poughkeepsie. The state RERP shows that there are over 180,000 civil defense dosimeters statewide stored by the Office of Disaster Preparedness (State RERP, Part III, § II, p. 17) FEMA has approved the use of civil defense dosimeters for radiological emergencies while more advanced types are phased in. (Davidoff, Tr. 11435-11436; Curran, Tr. 5072)

3.1-121 A number of witnesses contended that there were not enough low-range dosimeters in the hands of the agencies which they represent. (E.g., Guido, Tr. 4914-4915; Curran, Tr. 5045) None of these witnesses showed, however, that the approach adopted by REPG and FEMA, that is the use of a mix of different types of dosimeters while additional stocks of low-range self-reading devices are purchased, is proscribed by applicable federal planning standards. In addition, the high range dosimeters are appropriate for permitting workers to determine whether they have reached doses suggested in EPA's guidelines for emergency workers. (Curran, Tr. 5074-5075)

3.1-122 The December 16, 1982 Update Report concluded that the deficiencies in this planning standard noted in the Interim Findings had been adequately addressed. It noted that the state had agreed to initiate immediately the procurement of permanent record dosimeters and had started a phased acquisition of low-range self-reading dosimeters. The Update Report noted the high range civil defense dosimeters would be used in the interim and referred to other changes agreed to by the State. (December 16, 1982 Update Report, p. 7)

3.1-123 The 1983 Post Exercise Assessment found that radiological exposure control procedures at the state's Personnel Monitoring Center during the March 9, 1983 exercise were acceptable. (1983 Post Exercise Assessment, p. 21) Overall procedures for controlling radiological exposure of emergency workers in Westchester County were found to be acceptable. (Id., pp. 32-33) In Rockland County, exposure control measures were found to be generally acceptable. (Id., p. 43) In Orange County, radiological control procedures were found to be acceptable. (Id., pp. 50-51) In Putnam County it was found that field monitoring personnel were equipped with dosimeters and were trained in their use. FEMA recommended provision of additional dosimeters and training for bus drivers and police officers. (Id., pp. 57-58)

3.1-124 Based upon its review of the record in this proceeding the Board finds that the State and County RERPs meet the requirements of 10 CFR § 50.47(b)(11).

10 CFR § 50.47(b)(12)

3.1-125 10 CFR § 50.47(12) and NUREG-0654 Planning Standard L read as follows:

> Arrangements are made for medical services for contaminated injured

individuals.

3.1-126 On April 4, 1983, the Commission issued an order interpreting this regulation. (See Southern California Edison Co., (San Onofre Nuclear Generating Station Units 2 and 3), CLI-83-10) The Commission decided that with regard to "contaminated injured individuals" i.e., individuals who are both injured and contaminated, emergency plans should provide information with regard to medical facilities' capabilities to treat these individuals. Arrangements for treating onsite personnel and emergency workers should be adequate to treat members of the general public who are injured and contaminated. With regard to individuals who are exposed to dangerous levels of radiation, the Commission found that a lesser degree of planning is necessary and that this care can be provided on an "as needed" basis during an emergency. (CLI-83-10, pp. 2-3) With regard to persons in this latter group the person is unlikely to need immediate medical care. (Id., p. 12)

3.1-127 Licensees have entered into an agreement with Peekskill Community Hospital to accept injured and contaminated personnel from the site. The personnel at the hospital are regularly trained in techniques of dealing

with injured and contaminated patients. Northern Westchester Hospital Center has agreed to serve as a backup hospital. In addition, Con Edison has retained a medical consultant from Brookhaven National Laboratory Hospital to supervise special cases. (Con Edison Emergency Plan §§ 6.5.2 to 6.5.3, and Appendix A; PA Ex. 43, Emergency Plan Procedures, Document IP-1021). In addition to the hospitals with which Licensees have agreements, other hospitals in the Indian Point area have facilities for treating contaminated injured individuals. (Murphy Testimony, pp. 6, 9) The State RERP has identified 47 hospitals throughout the state, many of them within close proximity to Indian Point, which have indicated that they have the capability for treatment of contaminated individuals. (State RERP, Part III, § II, pp. 12 to 12(c))

3.1-128 The 1983 Post Exercise Assessment found that the simulations to test compliance with this standard went well. (1983 Post Exercise Assessment, pp. 10, 32, 44) No deviations from the NUREG-0654 evaluation criteria were noted in the Post Exercise Assessment.

3.1-129 James Murphy offered testimony on behalf of UCS/NYPIRG in order to show that the offsite emergency

plans for Indian Point are not in compliance with NUREG-0654 Planning Standard L. (Murphy Supplemental Testimony, p. 1) Mr. Murphy presented the results of a survey he conducted of hospital emergency facilities, ambulance services and residential care facilities in the plume exposure EPZ. The testimony is premised on the assumption that the RERPs must provide for medical services for large numbers of the general public. As the Board recognized when it permitted the testimony to be introduced, (Tr. 11059-11060), the relevance of Mr. Murphy's testimony hinged for a large part on how the Commission would rule in the San Onofre case. The Board finds that the Murphy study is premised on a reading of 10 CFR § 50.47(b)(12) which is inconsistent with the Commission's ruling in San Onofre.

3.1-130 In addition, the Board finds that the Murphy Testimony is seriously flawed. Mr. Murphy is not a health physicist. Mr. Murphy's training in areas relevant to radiological health is extremely limited. (Tr. 11061-11062) This testimony presents the results of his first survey. (Tr. 11062) Mr. Murphy chose not to obtain a copy of New York State's survey of health facilities that can provide medical services to injured, contaminated

individuals. (Tr. 11086-11087) In addition, the survey suffers from other infirmities. Mr. Murphy did not weigh the responses to reflect different capabilities of different facilities. (Tr. 11081) Thus, a large facility which has the capacity to treat many injured contaminated individuals was treated the same as a very small facility. (Tr. 11081-11082) In addition, though Mr. Murphy acknowledged that facilities beyond the 10 mile EPZ would be available for use in a radiological emergency, his survey did not assess this capability. (Tr. 11091) Mr. Murphy also acknowledged that initial decontamination, the only capability of hospitals assessed in the survey, could be performed in facilities other than health care facilities. (Tr. 11091-11092) There was also imprecision in some of the survey questions. (See, e.g., Tr. 11076-11078, 11079-11080) The percentage of responses received by Mr. Murphy was distressingly small. Only 11 of 26 hospitals and 9 of 37 residential health care facilities responded. Sixteen responses, only 5 of which were written, were obtained from 96 ambulance surveys mailed. (Murphy Supplemental Testimony, pp. 15-16, 27-29, 38-40)

3.1-131. The Board finds that the State and County RERPs meet the requirements of 10 CFR § 50.47(b)(12).

10 CFR § 50.47(b)(13)

3.1-132 10 CFR § 50.47(b)(13) and NUREG-0654 Planning Standard M read as follows:

> General plans for recovery and reentry are developed.

3.1-133 Both the State RERP (see, e.g., Part II, IV, pp. IV-1 to IV-7) and the County RERPs, (see, e.g., Westchester RERP, § IV, attachment I to Procedure I-9), provide guidance on conducting necessary recovery and reentry operations.

3.1-134 The 1983 Post Exercise Assessment found that the State decision-makers at the Office of Disaster Preparedness Southern District EOC and State EOC presented a method by which the recovery of the contaminated area would be accomplished. (1983 Post Exercise Assessment, p. 21) The performance of recovery and reentry procedures in the four counties was rated as either good or acceptable. (Id., pp .33, 45, 52, 59)

3.1-135 The Board finds that the State and County RERPs meet the requirements of 10 CFR § 50.47(b)(13).

10 CFR § 50.47(b)(14)

3.1-136 10 CFR § 50.47(b)(14) and NUREG-0654 Planning Exercise Standard N read as follows:

> Periodic exercises are (will be) conducted to evaluate major portions of the emergency response capabilities, periodic drills are (will be) conducted to develop and maintain key skills, and deficiencies identified as a result of exercises or drills are (will be) corrected.

3.1-137 It will be seen that this planning standard requires that emergency plans provide for periodic exercises and drills as well as providing for the correction of deficiencies identified in the exercises and drills.

3.1-138 The RERPs provide for an annual exercise of a major portion of the state and counties' emergency response organizations in order to verify the capability to respond to an accident scenario requiring a response, as well as providing for periodic drills and exercise of different facets of emergency response and for the evaluation of corrective action to correct deficiencies identified in the exercises and drills. (See, e.g., State RERP, pp .II-12 to II-14, Part III, § I, Procedure F; Westchester RERP, § II (B)(3), Procedure 12)

3.1-139 The 1983 Post Exercise Assessment commented favorably on the relevance of the exercise. (pp. 21, 24, 33, 45, 52, 59, 61 and 63)

3.1-140 The capacity of the State and four counties to respond to deficiencies revealed in exercises has been clearly demonstrated. This is obvious from an examination of Table I of the April 14, 1983 Post Exercise Assessment, the December 16, 1982 Update Report, the extensive revisions to the State and County RERPs that were made subsequent to the March 3, 1982 exercise and from the testimony of witnesses in this proceeding. (See, e.g., O'Rourke Testimony, p. 4; Davidoff/Czech Supplemental Testimony, pp. 2-3; Scalpi, Schmer, Tr. 12101-12105)

3.1-141 The Board finds that the State and County RERPs meet the requirements of 10 CFR § 50.47(b)(14). The plans provide for the required exercises and drills and for the adoption of corrective actions following drills and capacity to implement these provisions has been shown.

10 CFR § 50.54(b)(15)

3.1-142 10 CFR § 50.47(b)(15) and NUREG-0654 Planning Standard O read as follows:

Radiological emergency response training is provided to those who may be called on to assist in an emergency.

3.1-143 For the most part the RERPs rely upon organizations to perform the same duties during a radiological emergency which they perform during a non-radiological emergency. That is to say policemen and others with responsibilities under the RERPs perform duties that are, for the most part, the same duties they are prepared to perform on a day-to-day basis or during a non-radiological emergency. (Parsons Supplemental Testimony, pp.2-3) This fact is basic to the RERP. Thus, Procedure F to the State RERP, entitled Training Drills and Exercises," notes that functions that are normal for the agency's usual role, e.g., teaching a police officer to direct traffic are not provided for in the RERPs but rather only that training that is "unique to a radiological emergency." (State RERP, p. F-1)

3.1-144 In the 1981 and 1982 period Licensees and consultants retained by Licensees undertook a comprehensive training program of offsite personnel with emergency response duties in order to familiarize them with that portion of their duties which are unique to radiological emergency response. (Davidoff, Tr. 11455-11457) Since

that time responsibility for training of emergency workers has been assumed by the State. (Davidoff, Tr. 11457) New York State Ex. 9 represents the training provided by the State in the March 1982 to February 1983 period. The Exhibit shows a total of 4,892 attendees at state sponsored training. This training is scheduled to continue. (Davidoff, Tr. 11486) The key state and County personnel with emergency response duties such as the police have been trained. (Davidoff, Tr. 11397) In addition, training of radiological emergency workers in Orange and Putnam Counties has been extensive and has been actively supported by the State and licensees. (Schmer, Tr. 12149-12150; Scalpi, Tr. 12069). The State and the Licensees have regularly been willing to provide radiological emergency response training upon request. (Scalpi, Tr. 12099)

3.1-145. Although much has been done to provide radiological emergency training, not all emergency workers had yet received this training at the time the record of this proceeding closed (O'Rourke Testimony, p. 3; Davidoff/Czech Supplemental Testimony, p. 4) In Rockland County the pace of this training was adversely affected by the resistance on the part of the potential recipients to this training. (Davidoff, Tr. 11317) As Mr. Davidoff

noted, training is a constant, on-going process. (Tr. 11396) This process appears furthest developed in Putnam and Orange Counties and least developed in Rockland. In Westchester and Rockland, the pace of training has quickened.

3.1-146 The 1983 Post Exercise Assessment did not indicate any deviation from any of the NUREG-0654 criteria related to this Planning Standard, although there were recommendations for additional training in particular areas under evaluation criteria related to other planning standards.

3.1-147 The Board finds that the state and County RERPs meet the requirements of 10 CFR § 50.47 (b)(15).

10 CFR § 50.47(b)(16)

3.1-148 10 CFR § 50.47(b)(16) and NUREG-0654 Planning Standard P read as follows:

> Responsibilities for plan development and review and for distribution of emergency plans are established, and planners are properly trained.

3.1-149 The State REPG has been given overall responsibility for radiological planning in the state,

including liaison with other state agencies, local governments, federal agencies and the licensees (Davidoff/Czech Testimony, p .7) Mr. Davidoff and Mr. Czech are well qualified to perform their responsibilities with the REPG. Similarly, the various county personnel with responsibility for planning for radiological emergencies are well-trained to perform their responsibilities. (Marasco, Tr. 5491; McGuire/Hastings Testimony, pp. 1-3; Schmer, Tr. 12069; Scalpi, Tr. 12070) In addition, the RERPs provide for the distribution and revision of the plans on a regular basis. (See, e.g., State RERP, Procedure A-1; Putnam RERP, Procedure 11; WBCA Ex. 33, "Guidelines for Updating the Evacuation Portions of the Indian Point Radiological Emergency Response Plans and Procedures")

3.1-150 In its July 30 Interim Findings, FEMA found compliance with this Planning Standard to be significantly deficient. This finding was based upon concerns raised by Resolution No. 320 of the Rockland County Legislature. (December 16 Update Report, pp. 7-8) The Resolution, (Rockland Ex. 1), provided that the County would no longer participate with FEMA, NRC or Licensees in the development of a federally mandated RERP and would

prepare its own Nuclear Emergency and Preparedness Plan, as well as its own General Disaster Preparedness Plan. The Resolution provided, however, that, in the event of an actual radiological emergency, the County would cooperate fully with all agencies.

3.1-151 On December 7, 1982 the Rockland Legislature passed Resolution No. 829. This Resolution disavowed the existing Rockland RERP and directed the Chairman of the Legislature to obtain assistance from the State, FEMA, NRC and Licensees in developing the County's own plan. (December 16, 1982 Update Report, p. 8) The effect of this resolution was to make it somewhat easier for the State and concerned County personnel to work together to develop a workable emergency response capability. (Davidoff, Tr. 11473)

3.1-152 In response to FEMA's concerns related to the Rockland County issue, New York State decided to modify the State RERP to provide for state compensating measures if the County failed to respond to an emergency. (NYS Ex. 7, p. 9) The State RERP was in fact modified to provide for this possibility on a generic basis. (See, e.g., State RERP, I. I-11, as added by New York State Exhibit 6)

3.1-153 In its December 16, 1982 Update Report FEMA concluded that there remained a significant deficiency in this planning standard due to the Rockland County issue. The Update Report reached this conclusion despite noting that the State had "undertaken the developing of plans and procedures whereby the state officials would assume the County's responsibilities", (December 16, 1982 Update Report, p. 7), and despite "significant progress" in Rockland County related to funds, equipment, training and the development of a new plan by the County. (Id., p. 9) The primary reason for this finding was the "County's earlier lack of involvement in the planning and preparedness activities." (Id., p. 8) The Report stated that [u]ntil Rockland County's plan is developed, reviewed by the Regional Assistance Committee and personnel trained on the new plan, the original emergency plan must be utilized . . . . This plan has been disavowed by the Legislature of Rockland County. The Chairman of the County Legislature has also expressed to the NRC that the assumption of County functions by state personnel would not be adequate." (Id.) The Update Report did not attempt to assess the adequacy of a state response. Rather, it appears that FEMA concluded that compliance with this planning standard could be had only if Rockland were to develop a plan of its own.

3.1-154 Despite Rockland County's concerns about emergency planning at Indian Point, the State worked actively with County personnel to develop a plan which would be acceptable to the County. (Davidoff, Tr. 11362) While the initial drafting was proceeding, it became clear that the plan would not be completed in time for the March 1983 exercise at Indian Point. The State Disaster Preparedness Commission then directed that the plan which had been drafted by Rockland County personnel be adopted as a compensating measure to be used if a radiological emergency developed before the County completed its own plan. (Davidoff, Tr. 11361-11362) The Rockland County draft plan, as modified somewhat by REPG, is New York State Exhibit 10, the State's Compensating Measures for Rockland County. This was done with the consent and knowledge of Rockland County. (Davidoff, Tr. 11362-11363) The action by the State was pursuant to Article 2-B of the New York State Executive Law which permits the Governor to assume special powers in the event of a radiological emergency. (Davidoff, Tr. 11365)

3.1-155 Rockland County personnel participated in the March 9, 1983 exercise until 9:00 a.m., (McIntire, Tr.

14874), and did not perform badly. (McIntire, Tr. 14877) Thereafter, state personnel implemented the state compensating measures by utilizing state personnel. Mr. Davidoff testified that these personnel performed well. (Tr. 11478) FEMA agreed that their performance was generally good. (McIntire, Tr. 14871)

3.1-156 Despite a generally favorable review of how the March 9, 1983 emergency exercise was actually conducted in Rockland County, the 1983 Post Exercise Assessment found the response deficient. FEMA reached this conclusion for two reasons. First, the County had not finalized and adopted a RERP and did not participate in the exercise. FEMA cited its proposed rule, 44 CFR Part 350, for the proposition that the concept of Emergency Planning Zones implies mutually supportive efforts by federal, state and local governments. (1983 Post Exercise Assessment, p. 35) Thus, FEMA believed that participation in emergency planning by all these levels of government was required to satisfy federal planning standards, and that FEMA could not form a judgment of the adequacy of emergency preparedness in Rockland County until a County plan had been developed and exercised. (McIntire, Tr. 14920) Second, FEMA concluded that since the State RERP contemplated the State

supplementing county personnel and resources, rather than supplanting them, the State had failed to follow its own plan during the exercise. (Id., p. 36)

3.1-157 The Board finds the two reasons cited by FEMA in its 1983 Post Exercise Assessment for its finding of inadequacy to be without sufficient basis. With regard to the first rationale, although the Board agrees that cooperation between all levels of government would be the ideal way of approaching radiological emergency planning, a local government like Rockland County cannot be given a veto over this planning effort simply by choosing not to participate. In addition, FEMA itself has acknowledged that state resources could compensate for the failure of a locality to participate in planning a locality (PA Ex. 55; McIntire, Tr. 14922-14923). With regard to the second reason offered, i.e., that the State RERP contemplated a state effort to supplement and not supplant the County, FEMA has too narrowly read the State RERP. FEMA cites two sections of the RERP which state that in those instances in which a county cannot or does not implement its RERP the state will implement the county's RERP "using state and local resources and personnel." Since FEMA believes only state resources and personnel were utilized in Rockland on

March 9, 1983, it concluded that the State violated its own RERP. Page I-11 of the State RERP permits the state to "take the necessary actions to respond" if a county does not act. In addition, we do not read the use of the connective "and" as requiring the use of both state and local personnel in each instance of a county's inability to act. In any event, Rockland personnel did participate in the exercise, (see Finding 3.1-155), and the resources of the County, (e.g. the County EOC), were utilized.

3.1-158 During his cross-examination, Mr. McIntire offered an additional reason for FEMA's finding with regard to Rockland County to wit, concern that despite its good performance during the March 1983 exercise, the State might not have the resources to respond to an actual emergency. (Tr. 14871) FEMA itself agreed that in an actual emergency there would be full participation by County personnel and a full commitment of County resources. (McIntire, Tr. 14879)

3.1-159 In considering the actions of Rockland County it is important to note that the County's Legislature has sought the immediate shutdown of the Indian Point units (Gdanski Supplemental Testimony, p. 1, Gdanski

Supplemental Testimony, Ex. C) The Chairman of the Citizens Advisory Committee which is to review the draft of the County's plan is Judith Kessler, (I), McGuire, Tr. 10953), a representative of the Rockland Citizens for Safe Energy, an intervenor group in the proceeding which has been extremely critical of Indian Point emergency planning. The Chairman of the schools subcommittee, Bernard Flicker, is president of Rockland Families to Close Indian Point. (Flicker Testimony, p. 1; Flicker, Tr. 11258) Finalization of the County's draft plan must await review by the Citizens Advisory Committee and final approval by the County Legislature. (D. McGuire, Tr. 10945) Despite the County's expressed concern about emergency planning, its efforts to develop its own plan has not proceeded in a speedy manner. The Citizens Advisory Committee has been meeting only one evening a week to review the draft plan. (Reisman, Tr. 11045) Although Resolution No. 320 directed that a plan be presented for adoption by the Rockland County Legislature by December 31, 1982, (Rockland Ex. 1, pp. 2-3), the Chairman of the County Legislature refused to fix a date for adoption but estimated that it would be 1984 before the County's own plan is in place. (Reisman, Tr. 11045) We find that the County's efforts in furtherance of radiological emergency

planning have not been commensurate with the County leadership's own perceptions of the risk from Indian Point operation which have presumably motivated them.

3.1-160 The State REPC has found that emergency planning at Indian Point is in substantial compliance with each of the sixteen planning federal standards. The only serious question with regard to compliance with this planning standard relates to the Rockland County issue. Based upon our review of the record the Board finds that despite the difficulties encountered to date in arriving at a plan which would be acceptable to the County, the RERPs are in compliance with this standard. We find that the compensating measures adopted by the State provide adequate assurance that the public health in the County would be protected in the extremely unlikely event that there were to be a radiological emergency at Indian Point before the County adopts its own radiological emergency response plan and that upon the occurrence of such an event the County failed to provide its own personnel and resources. We find, however, that the uncontradicted evidence in the proceeding is that the County would fully commit its emergency workers and equipment.

B. LICENSEES' EMERGENCY PLANS

3.1-161 Onsite emergency planning for Indian Point Unit No. 2 is contained in Con Edison's Emergency Plan, which describes the responsibilities of Con Edison personnel at the Indian Point site during both radiological and non-radiological emergency situations, and the Emergency Procedures Document, containing procedures for implementing the Emergency Plan. (Con Edison Onsite Testimony, p. 6) Onsite emergency planning for Indian Point Unit No. 3 is contained in the Power Authority's Emergency Plan and Emergency Planning Implementation Procedures Document. (Power Authority Onsite Testimony, pp. 12-13)

3.1-162 Con Edison and the Power Authority have entered into a Memorandum of Understanding to provide for joint planning and implementation of emergency planning activities at Indian Point. (Con Edison Onsite Testimony, pp. 6-7; Power Authority Onsite Testimony, p. 14)

3.1-163 There is considerable duplication between the requirements of 10 CFR § 50.47(b) and Appendix E to 10 CFR Part 50. That is to say that subjects such as

organization, notification and communication are covered by both 10 CFR § 50.47(b) and Appendix E. The Sears Testimony is the only attempt to assess the level of compliance of Licensees' plans with all of these requirements. Mr. Sears addressed each of the 16 parts of 10 CFR § 50.47 in sequence. In deciding whether any of these requirements are met Mr. Sears would also answer with regard to those portions of Appendix E that dealt with the same subject as a particular part of 10 CFR § 50.47(b). The Board will follow Mr. Sears' approach.

10 CFR § 50.47(b)(1)

3.1-164 This regulation deals with the assignment of responsibility for emergency response, including the delineation of responsibility among different organizations and the provision of staff to respond on a continuous basis. This subject is also dealt with by Appendix E.I.V.A.

3.1-165 The Emergency Plans for Indian Point Units 2 and 3 provide a full description of the Onsite Emergency Organizations to deal with emergencies. The plans include a definition of authorities, duties, responsibilities and the means of notifying individuals

with responsibilities under the respective plans. (Con Edison Onsite Testimony, p. 6; Power Authority Onsite Testimony, p. 3; Sears Testimony, pp. 2-3)

3.1-166 Emergency Plans for Indian Point Units 2 and 3 identify various off-site agencies and organizations which will be part of the overall response organization. (Sears Testimony p. 3) as well as describing the responsibilities of various onsite support organizations. (Id., pp. 3-4)

3.1-167 Licensees' emergency plans provide a delineation of responsibilities among response organizations, (Id., p. 4), set forth the duties and responsibilities of individuals who would direct emergency response, and provide well-defined staff assignments and a description of the responsibilities of an Emergency Director. (Id., p. 4; Con. Edison Onsite Testimony, p. 6; Power Authority Onsite Testimony, pp. 14-15)

3.1-168 The Emergency Plans for Indian Point 2 and 3 provide for staffing on a 24-hour basis and provide the means to respond to an emergency of a continuous basis. (Sears Testimony, p. 5). Provisions exist in the

plans for both units to obtain support from each organization's corporate staffs for initial and long term response. (Id., p. 5; Jackson, Tr. 14760-14761)

3.1-169 The Emergency Plans for both units contain adequate written agreements with federal, State, local and other support organizations. (Sears Testimony, pp. 5-6)

3.1-170 Staff testified that the emergency plans for both units 2 and 3 meet the requirements of 10 CFR § 50.47(b)(1), as well as those portions of 10 CFR 50 Appendix E.IV.A that address the organizational issues dealt with by 10 CFR § 50.47(b)(1). (Id., p. 6) Based upon our review of the record in this proceeding we agree with the Staff conclusion.

10 CFR § 50.47(b)(2)

3.1-171 10 CFR § 50.47(b)(2) and NUREG-0654 Planning Standard B read as follow:

> On-shift facility licensee responsibilities for emergency response are unambiguously defined adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, timely augmentation of response capabilities is available and the interfaces among various on-site response activities and offsite

support and response activities are specified.

3.1-172 The regulation deals with the requirements of defining responsibility and providing staff for the onsite organizations, as well as providing for augmentation and support from offsite organizations. Those provisions of Appendix E.IV.A that are relevant to this regulation will also be considered.

3.1-173 Both the Con Edison and Power Authority plans provide for staffing of an emergency organization on a 24-hour basis, (Sears Testimony, p. 6), designation of an Emergency Director with responsibilities corresponding to an Emergency Coordinator and an established line of succession to this position. (Id., pp. 6-7)

3.1-174 Clear definition of on-shift personnel responsibilities are provided for in the emergency plans of both licensees. Provisions are made for initial staffing of positions, as well as for augmenting of emergency staff. (Id., pp. 7-8) The emergency plans give the minimum staffing for required on-shift personnel and additional staffing within 60 minutes and thus meet the goals of the time response called for by the Table B-1 of

NUREG-0654. (Id., p. 8; Con Edison Onsite Testimony, pp. 15-16; Power Authority Onsite Testimony, pp. 15,17)

3.1-175 Licensees' plans provide for interfaces among the various onsite emergency response activities and off-site support and response activities. The plans also identify contractors and private organizations that may be called upon for assistance in an emergency. (Sears Testimony, pp. 8-9; Con Edison Onsite Testimony, pp. 6, 12)

3.1-176 The plans of Con Edison and the Power Authority identify by position and function the personnel within the licensee organization who are to make offsite dose projections and transmit these projections to appropriate governmental authorities. (See Con Edison Emergency Plan § 5.2)

3.1-177 Staff has concluded that the emergency plans for Indian Point units 2 and 3 meet the requirements of 10 CFR § 50.47(b)(2), as well as those of 10 CFR Appendix E.I.V.A that are relevant to this regulation. (Sears Testimony p. 10) Based upon review of the record we concur with Staff's finding.

10 CFR § 50.47(b)(3)

3.1-178 This planning standard deals with provisions for using the assistance resources from non-licensee sources, accommodating state and local staff at licensees' near-site EOF and identifying sources capable of augmenting the emergency response. Those portions of Appendix E.I.V.A relevant to the standard (i.e. E.IV A (7) and (8)) will also be considered.

3.1-179 The emergency plans of both licensees provide for obtaining additional resources from federal, state and local agencies, as well as from other organizations. (Con Edison Testimony p. 12; Brons Tr. 11697; Sears Testimony pp. 5, 13) Both licensees have made arrangements to participate in the Federal Radiological Monitoring and Assessment Program. (Sears Testimony, p.10)

3.1-180 Both Licensees' plans identify the State and local authorities who are responsible for ordering protective actions. (See, Section 6.0 of both Licensees' plans)

3.1-181 Both Licensees' plans provide for sending a Licensee representative to the offsite EOC, (Sears

Testimony, pp. 10-11), identifying radiological laboratories which can be used to provide monitoring and analysis services, (Id., p. 11), and identify organizations other than federal, State and local agencies which can provide assistance. (Id., pp. 9-11)

3.1-182 Both Licensees' plans provide for accommodating State and local personnel at the near-site EOF. (Id., p. 11)

3.1-183 Estimated times of arrival of Federal resources and the availability of laboratories to provide monitoring and analyses service have been identified. (Id., p. 12)

3.1-184 Staff concluded that the provisions of Licensees' emergency plans meet the requirements of 10 CFR § 50.47(b)(3) as well as those of Appendix E.I.V.A of 10 CFR Part 50 which are related to this regulation. (Id., p. 12) The Board concurs in this judgment.

10 CFR § 50.47(b)(4)

3.1-185 This planning standard, to the extent it deals with Licensee emergency planning, requires the

establishment and use of an emergency classification and action level system. Those parts of E.IV. B and C related to this requirement will also be considered.

3.1-186 Section 4 of the Emergency Plans of both Licensees and their respective implementing procedures provide methods and tests for initiating conditions and emergency action levels to be used in the evaluation of which emergency classification must be declared (i.e. Notification of Unusual Event, Alert, Site Emergency and General Emergency). Tables in the implementing procedures specify plant conditions which are used in declaring an emergency and notifying governmental agencies. (Con Edison Onsite Testimony, p. 9; Power Authority Onsite Testimony, pp. 12-13, Sears Testimony, pp. 12-13) The emergency plans and implementing of procedures are consistent with Appendix I of NUREG-0654. (Sears Testimony, pp. 13-14)

3.1-187 Emergency Action Levels (EALs) established by Licensees' emergency plans are usable in determining the timing and type of protective measures to be considered within and outside the site boundary. (Con Edison Onsite Testimony, pp. 8-10; Power Authority Onsite Testimony, pp. 11-14) PASNY Procedure IP-1017 and Con

Edison Procedure IP-1013 "Recommendations of Protective Action for Off-site Population," describe the method to be used by Licensees in determining the protective actions to be recommended to offsite authorities. These procedures are based upon Environmental Protection Agency Protective Action Guides and on plant conditions described in the EALs. (Sears Testimony, p. 35) The Licensees' plans also provide for protective response actions for onsite personnel. (Id., pp. 34, 36)

3.1-188 The Staff has concluded that the emergency plans of both Licensees meet the requirements of 10 CFR § 50.47(b)(4) as well as the requirement of Appendix E.IV B and C that are related to the requirements of the regulations. (Sears Testimony, p. 14) The Board concurs in this judgment.

10 CFR § 50.47(b)(5)

3.1-189 This regulation, to the extent it applies to Licensees, requires procedures for notification of state and local organizations and of emergency personnel, establishment of initial and follow-up communications and the means to provide early notification and instruction to the public within the plum exposure EPZ. Compliance of Licensees' emergency plans with those portions of Appendix E.IV C and D which deal with the same subjects will also be considered.

3.1-190 Con Edison Procedure IP-1002 and Power Authority Procedure IP-1030 describe the initial and follow-up notification to federal, State, local and Licensee offsite emergency organizations. (Sears Testimony, p. 14) These procedures include initial and follow-up communications which contain relevant information on radioactive release, proposed protective actions and meteorological conditions. The procedures permit the prompt and accurate notification of State and local response organizations. (Id., p. 15)

3.1-191 Licensees' emergency procedures provide for communications to Licensee emergency response

personnel. (Id., p.15) Con Edison has a page/party communications system throughout Indian Point 2, as well as a conventional telephone system. Radio communications to field personnel, onsite security forces, corporate headquarters as well as to governmental authorities is provided. The Power Authority also has an extensive onsite communications system. (Con Edison Onsite Testimony, pp. 13-14; Power Authority Onsite Testimony, pp. 27-28; Sears Testimony, pp. 18-19)

3.1-192 Licensees have made provisions with regard to the initial and follow-up messages from the plant to offsite authorities as well as the means for providing supporting information to offsite authorities for messages intended for the public. (Sears Testimony, pp. 15-17)

3.1-193 The recently-enhanced Alert and Notification System, together with the EBS system, provide the means for the prompt notification and instruction of the public within the plume exposure pathway. (Sears, Tr. 12326-12377; Con Edison Onsite Testimony, pp. 13, 15; Con Edison Onsite Supplemental Testimony, p. 1; Jackson, Tr. 11759)*

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* See, also Findings 3.1-44 to 3.1-45.

3.1-194 In its prepared testimony Staff concluded that Licensees' emergency plans met the requirements of 10 CFR § 50.47(b)(5) as well as those requirements of Appendix C and D which are related to the same subjects, with the exception that, due to deficiencies in the siren system noted in the March 1982 exercise, it could not conclude that Appendix E.IV. D(3) was met. (Sears Testimony, pp. 15, 17, 18) Staff modified this testimony relating to the ANS siren system to note the successful operation of the siren and EBS system during the March 9, 1983 exercise and the prompt notification of state and local officials. (Sears, Tr. 12240-12241) Based upon its review of the record in this proceeding, the Board concludes that Licensees' emergency plans meet the requirements of 10 CFR § 50.47(b)(5), as well as those of Appendix E.IV. C and D that are related to these requirements.

10 CFR § 50.47(b)(6)

3.1-195 This regulation deals with the provision for prompt communications among principal response organizations, to emergency personnel and to the public. Portions of Appendix E. IV. C and E are relevant to this subject.

3.1-196 Section 7.2 of the Con Edison Emergency Plan and Section 7.3 of the Power Authority plan describe the means of implementing prompt communications with offsite personnel. (Sears Testimony, p. 19) The communication system includes a Radiological Emergency Communication System (RECS) which provides immediate communication between the reactor control rooms, emergency response facilities and State and county warning points. (Sears Testimony, p. 19, Con Edison Onsite Testimony, pp. 12-13) In addition, the communication system for both Licensees includes a public address paging system, personal radio (beeper) paging system, audible alarms, direct line telephones, a radio system, a NAWAS (National Warning System) line to the County and State warning points and to the City of Peekskill, the NRC Emergency Notification System and the NRC Health Physics Network. (Sears Testimony, p. 19)

3.1-197 Licensees have provided for manning communication links on a 24-hour basis. (Id.) Provisions have been made for communicating between the units and the near-site EOF, EOC's, radiological monitoring teams, and fixed and mobile medical support facilities. (Id., pp. 19-20) Licensees' plans provide the means for activating

its response personnel and for periodic tests and drills of communication equipment. (Id., pp. 20-21)

3.1-198 Staff concluded that the plans of Licensees meet the requirements of 10 CFR § 50.47(b)(6) as well as those requirements of Appendix E.IV. C and E which are related to communications. (Sears Testimony, p. 21) We concur in that judgment.

10 CFR § 50.47(b)(7)

3.1-199 This regulation deals with public information and education. Appendix E.IV. D 2 also relates to the periodic dissemination of information to the public.

3.1-200 A public information booklet has been developed and mailed to the public within the EPZ. This booklet provides that information on initial notification and protective actions the public would be directed to take in case of radiological emergency be revised annually. (Sears Testimony, pp. 21-22; Con Edison Onsite Testimony, pp. 14-15) In addition, telephone directory inserts, posters and radio announcements have been developed to provide information to transients and to others who may not

have received the pamphlet. (Sears Testimony, p. 2?)*

3.1-201 Licensees have developed designated points of contact with the media as well as procedures for the dissemination of information to the public and the media. (Id., p. 22) Licensees' plans have also made provision for control of rumors. (Id., p. 23)

3.1-202 Staff concluded that Licensees' plans meet the requirements of 10 CFR § 50.47(b)(7) as well as those requirements of Appendix E.IV. D related to public information. (Id., p. 23) We concur in that judgment.

10 CFR § 50.47(b)(8)

3.1-203 This regulation calls for the provision and maintenance of adequate emergency facilities and equipment to support emergency response. Portions of Appendix E.IV. E deal with the same subject of this regulation. In addition, to the extent that it deals with

* See, also Findings 3.1-66 to 3.1-67.

the maintenance of emergency equipment, Appendix E.IV. G relates to areas encompassed within this regulation.

3.1-204 Con Edison Procedures IP-1035 and IP-1023 and PASNY Procedures IP-1045 and IP-1047 describe the establishment and operation of each Licensee's Technical Support Center and Operational Support Center. In addition the plans provide for a joint Emergency Operation facility. (Sears Testimony, pp. 23-24) The Technical Support Centers and Emergency Operation Facilities were established in response to the Post-TMI requirements of NUREG-0737. (Id., p. 24) The procedures for the activation and operation of these facilities include the provision of equipment for monitoring and analysis of plant parameters and offsite conditions. Rosters of personnel which include the designation of where each person is to report are provided. (Id., pp. 24-25)

3.1-205 Both Licensees have procedures and equipment to be used for onsite monitoring for initiating emergency measures. The following items, together with personnel trained in their use, are provided for:

a) Survey instruments - e.g., ion chambers Geiger-Muller detectors.

b) Laboratory instruments - e.g., proportional counters.

c) Personnel dosimetry - e.g., pocket ion chambers, thermoluminescent dosimetry, and film badges.

d) Area monitors - i.e., fixed radiation detectors in the plants.

e) Process radiation monitors.

(Con Edison Onsite Testimony, p. 16, 21-22; Power Authority Onsite Testimony, pp. 21-22; Sears Testimony, p. 25)

3.1-206 Licensees' plans have also made provision for offsite monitoring. Section 7.3.2 of the Con Edison Emergency Plan and Section 7.4.2 of the Power Authority Plan describe facilities and equipment for offsite monitoring. Licensees have also installed a Reuter-Stokes Sentri 1011 Remote Environmental Radiation Monitoring System consisting of 16 monitors within a 2-mile radius of the plants with the capability to telemeter radiation and meteorological readings to a processor which transmits it to the Meteorological Information and Dose Assessment System (MIDAS) computer system at the EOF. In addition, Ludlum area radiation monitors are located at approximately 25 offsite locations which are manned on a 24-hour basis. Finally, the Con Edison Nuclear Environmental Monitoring Group conducts a continuous monitoring program for the site and surrounding area. (Con Edison Emergency Plan, § 7.3.2;

Con Edison Implementation Procedure IP-1015; Power Authority Emergency Plan § 7.4.2; Power Authority Procedure IP-1011; Power Authority Onsite Testimony, pp. 22-23; Con Edison Onsite Testimony; pp. 16-17; Sears Testimony, pp. 25-26)

3.1-207 Licensees have an extensive system for data acquisition and analysis of the meteorology of the Indian Point area. The meteorology of the area has been extensively studied both by consultants under contract to Licensees and Licensees' meteorologists. A primary 122 meter tower and two ten-meter backup towers, are located onsite. These towers have been instrumented in accordance with NRC Reg. Guide 1.23, "Meteorological Program in Support of Nuclear Power Plant," and NUREG-0654. (Con Edison Onsite Testimony, pp. 20-21; Con Edison Supplemental Onsite Testimony, p. 2) The 122 meter tower provides real-time meteorological data. Readouts of wind speed and direction and the ability to calculate Pasquill Stability Category are available in the control rooms. Access to this data is also available remotely by telephone data telemetry. This data is sent to the MIDAS and ARAC Systems in the EOF. The ten-meter towers provide real time wind speed and direction data and an estimate of the Pasquill

Category. This data is continuously recorded and sent to MIDAS. This data is available in the EOF and can also be accessed remotely by telephone. (Con Edison Procedure IP-1016; Power Authority Procedure IP-1003; Power Authority Onsite Testimony, pp. 23-24; Con Edison Onsite Testimony, p. 16; Sears Testimony, pp. 26, 51)

3.1-208 Both Licensees have procedures establishing the location, specification and frequency of inventory, operational and calibration checks for air sampling and counting equipment, portable survey instruments, dosimetry and respiratory equipment, anti-contamination clothing, procedures and maps. (Sears Testimony, pp. 26-27)

3.1-209 The Staff of the Commission has concluded that the plans of the Licensees comply with the requirements of 10 CFR § 50.47(b)(8) as well as those parts of Appendix E.IV. E and G of 10 CFR Part 50 that relate to the same subjects. (Id., p. 27) Based upon our review of the record, we concur in the finding.

10 CFR § 50.47(b)(9)

3.1-210 This regulation requires that licensee emergency plans provide for methods, systems and equipment to assess and monitor actual or potential offsite consequences of radiological emergency. Parts of Appendix E.IV.B and E relate to the same subject area.

3.1-211 Licensees have procedures for assessing and monitoring offsite consequences of a radiological release. (Sears Testimony, pp. 27-28) Licensees have the capability through the Meteorological Information Dose Acquisition System (MIDAS) and Atmospheric Release Advisory Capability (ARAC) to forecast plume movement and to make dose assessments. This equipment provides for a range of accident conditions and permits the rapid assessment of the magnitude and location of radiological releases. (Sears Testimony, pp. 27-29; Con Edison Onsite Testimony, pp. 23-24; Power Authority Onsite Testimony, pp. 26-27; see, also Findings 3.1-205 to 3.1-207)

3.1-212 Staff concluded that the Licensees' plans meet the requirements of 10 CFR § 50.47(b)(9) as well as those parts of Appendix E.IV.B and E which relate to assessing and monitoring potential offsite consequences.

(Sears Testimony, p. 29) Based upon our review of the record we concur in that finding.

10 CFR § 50.47(b)(10)

3.1-213 This regulation requires that licensee plans provide for a range of protective actions in response to an emergency and that guidance, consistent with Federal guidelines, be established for the selection of protective actions. Portions of Appendix E.IV.D and E of 10 CFR Part 50 also relate to licensee requirements regarding protective actions. Also, the introductory paragraph to Appendix E.IV.D is relevant to this regulation to the extent it requires licensees to supply an analysis of the time required to evacuate or to take other protective actions.

3.1-214 Licensees have developed the means for alerting persons onsite or persons in areas controlled by Licensees in the event of an emergency. Procedures have been established for evacuating onsite non-essential personnel, for accounting for individuals who are onsite during an emergency and for radiological monitoring and contamination surveys of people evacuated from the site. (Sears Testimony, pp. 32-34)

3.1-215 Licensees' procedures provide for protective equipment for individuals remaining or arriving onsite during an emergency, including protective clothing, respiratory devices and survey equipment. In addition, a large supply of potassium iodide is available which would be administered at the direction of the Emergency Director. (Id., p. 34)

3.1-216 Licensees' procedures describe the methods to be used to recommend evacuation or other protective actions to offsite authorities. The bases for the recommendations are the U.S. Environmental Protection Agency's Protective Action Guides and the plant conditions described in the Emergency Action Level Procedures. These bases are sufficient for the Licensees to make decisions on recommended protective actions. (Id., pp. 34-36)

3.1-217 Licensees have submitted and developed time estimates for evacuation and for taking other protective actions which meet the criteria of NUREG-0654 and therefore meet the requirements of Appendix E.IV relating to evacuation time estimates or estimates to take other protective actions. The evacuation time estimates are reflected in Licensees' emergency plans. (Id., pp. 29-32)

3.1-218 Staff concluded that Licensees' emergency plans meet the requirements of 10 CFR § 50.47(b)(10) as well as the requirements of Appendix E.IV.D and E which relate to protective actions for on-site personnel and the capability to recommend protective actions to offsite authorities. (Id., p.36) We concur in that judgment.

10 CFR § 50.47(b)(11)

3.1-219 The regulation requires provisions for controlling radiological exposure of emergency workers which are to be consistent with EPA Emergency Worker and Lifesaving Protective Actions Guides. In addition, Appendix E.IV.E (4) relates to decontamination of onsite individuals.

3.1-220 Licensees have established measures for controlling exposure of emergency workers to radiation which are consistent with EPA Protective Action Guides. The capability exists to determine doses received by emergency workers on a 24-hour basis. (Sears Testimony, pp. 36-37; Con Edison Onsite Testimony, p. 17; Power Authority Onsite Testimony, pp. 31-32)

3.1-221 Decontamination procedures and supplies and facilities to be used in decontaminating onsite individuals are set forth at Sections 6.4.3.1, 6.5.2 and 6.5.4 of Con Edison's Emergency Plan, in Con Edison Procedure IP-1008 and in Section 6.5.4 of the Power Authority's Emergency Plan.

3.1-222 Staff concluded that the emergency plans of Licensees comply with the requirements of 10 CFR § 50.47(b)(11) as well as both those portions of Appendix E.IV.E that deal with the same subject. (Sears Testimony, pp. 37-38) We concur in that finding.

10 CFR § 50.47(b)(12)

3.1-223 This regulation requires plans for providing medical services to contaminated injured individuals. Portions of Appendix E.IV.E also relate to the provision of medical aid to onsite individuals.

3.1-224 Provisions for onsite first aid capability include a First Aid and Decontamination Suite, an alternate area for treatment of radiation casualties at the Con Edison Medical Bureau Service Center at the EOF and three other onsite first aid rooms. (Sears Testimony, p. 38)

3.1-225 Provision, including agreements with an ambulance service, have been made for transporting contaminated injured individuals. In addition, Licensees have agreements with the Peekskill Community Hospital to accept onsite injured and injured contaminated individuals. (Id., pp. 38-39; Con Edison Onsite Testimony, p. 18)

3.1-226 Staff concluded that the emergency plans of Licensees comply with the requirements of 10 CFR § 50.47(b)(12) as well as those of Appendix E.IV.E. that involve the same subject area. (Sears Testimony, p. 39) The Board concurs in this finding.

10 CFR § 50.47(b)(13)

3.1-227 This regulation requires that licensees' plans provide for general plans for recovery and reentry. Appendix E.IV.H also relates to criteria to be used in determining when reentry to the facility or resumption of operations should be made.

3.1-228. Section 9 of each Licensee's emergency plan describes recovery operations, including the responsibilities of Licensees' Recovery Manager, reentry,

assessment, repair and return to operations. The plans provide that radiation limits in 10 CFR Part 20 will be observed. The plans also establish a method for periodically estimating total population exposure. (Sears Testimony, pp. 39-40)

3.1-229 Staff concluded that Licensees' emergency plans comply with the requirements of 10 CFR § 50.47(b)(13) as well as Appendix E.IV.H of 10 CFR Part 50. (Id., p.40) The Board concurs in that finding.

10 CFR § 50.47(b)(14)

3.1-230 This regulation requires that the plans provide for periodic exercises and drills and for the correction of deficiencies noted in these drills and exercises. Provisions for drills and exercises are also required by portions of Appendix E.IV.F.

3.1-231 The Licensees' plans (Con Edison Plan § 8.1.2 and Power Authority Plan Table 8-11) provide for periodic drills and exercises of various sorts (Staff Testimony, p. 43; Con Edison Onsite Testimony, p. 18).

3.1-232 On the basis of its evaluation of March 4, 1982 exercise at Indian Point 3, NRC Staff concluded that the Power Authority had shown overall capability to respond to an emergency and that Licensees' actions are adequate to protect the public. The evaluation found areas for improvement which the Licensee agreed to undertake. (Sears Testimony, p. 42)

3.1-233 In its March 29, 1983 Inspection Report of the March 9, 1983 exercise at Indian Point 2, (NRC Ex. 7, admitted Tr. 14369), NRC Staff found no violations and also found that Licensees' performance demonstrated the ability to implement its Emergency Plan and Implementing Procedures in a manner which would adequately protect the public. (NRC Ex. 7, p. 7) The Inspection Report noted areas for improvement which Con Edison agreed to undertake. (Id.; Sears, Tr. 14663-14664)

3.1-234 The NRC's Office of Inspection and Enforcement assures that problem areas identified during an exercise are corrected. (Sears Testimony, p. 42)

3.1-235 Licensees provide for internal critiques of drills and exercises and corrective actions to correct

deficiencies noted during drills and exercises. (Con Edison Onsite Testimony, p. 18; Power Authority Onsite Testimony, p. 32; Con Edison Emergency Plan, pp. 8-5 to 8-6)

3.1-236 Staff concluded that Licensees' provisions for conducting exercises and drills and for taking corrective actions meet the planning standard of 10 CFR § 50.47(b)(14), as well as the requirements of Appendix E.IV.F that relate to drills and exercises. (Sears Testimony, p. 44) The Board concurs in this finding.

10 CFR § 50.47(b)(15)

3.1-237 This regulation requires that radiological emergency response training be provided to those who may be called on to assist in an emergency. The subject of training of emergency response workers is also dealt with in portions of Appendix E.IV.F.

3.1-238 Each Licensee has developed an Emergency Planning Training Program which provides detailed lesson plans which are appropriate to particular emergency function. (Sears Testimony, p. 44) Con Edison plant

operators receive intensive training, including the use of an onsite simulator, which simulates actual plant operations, in order to assure the recognition of abnormal plan conditions. (Con Edison Onsite Testimony, p. 8) Annual radiological emergency response training is provided to all persons who may be called upon to assist in an emergency. Licensees also provide training to New York State Police, local fire departments, ambulance corps and designated hospitals which may be called upon to respond to an emergency. (Con Edison Onsite Testimony, pp. 18-19; Jackson, Liebler, Tr. 11739-11741; Brons, Tr. 11699) Con Edison has developed a table which is a matrix of course versus the appropriate persons to take such training. The matrix includes the categories identified in NUREG-0654 and radiological monitoring and dose assessment personnel taking such courses. The Power Authority has a similar system. (Sears Testimony p. 45) Section 8 of each Licensee plan includes tables that summarize the required training and its frequency and the personnel taking the training. In addition, each Licensee has a computerized system to insure that retraining is received when scheduled. (Sears Testimony, p.45)

3.1-239 Con Edison and the Power Authority provide an annual training program for news media personnel. (Con Edison Plan, pp. 8-10)

3.1-240 Staff has concluded that the provisions of Licensees' emergency plans related to training meet the requirements of 10 CFR § 50.47(b)(15) as well as those requirements of Appendix E.IV.F which relate to training. (Sears Testimony, pp. 45-46) Based upon its review of the record in this proceeding we concur in this finding.

10 CFR § 50.47(b)(16)

3.1-241 This regulation requires that plans establish responsibility for plan development, and for review and distribution of the plan and that planners be well-trained. Portions of Appendix E.IV.G relate to the maintenance of the emergency plan.

3.1-242 The plans of both Licensees have designated an employee for emergency plan development and have provided for the review and processing of possible changes to the plans. Procedures are in place for the distribution of the plans and of revisions to the plans. (Sears Testimony, pp. 46-47)

3.1-243 The Licensees' emergency response training program includes provisions for training of emergency planners. (Id., p. 47)

3.1-244 Con Edison's emergency plan provides for an annual audit of its emergency preparedness program by its Quality Assurance organization which provides an audit report to the Nuclear Facilities Safety Committee. The Power Authority provides for a similar audit. (Id., pp. 47-48)

3.1-245 The emergency plans of Licensees provide a list of supporting plans and their sources, an appendix listing implementing procedures and a table of contents. (Id., p. 48)

3.1-246 Staff concluded that Licensees' emergency plans meet the requirements of 10 CFR § 50.47(b)(16) as well as those parts of Appendix E.IV.G that relate to updating of emergency plans. (Id.) We concur in that finding.

C. ULTIMATE FINDING

3.1-247 The Board finds that the State and County RERPs and Licensees' emergency plans are in compliance with the planning standards of 10 CFR § 50.47(b) and that Licensees plans are in compliance with the standards of Appendix E to 10 CFR Part 50. Accordingly, Contention 3.1 is denied.

III. CONTENTION 3.2

3.2-1 Contention 3.2 reads as follows:

> The emergency plans for Indian Point Units 2 and 3 do not conform with NRC/FEMA guidelines because the assumptions made therein with respect to human response factors during a radiological emergency are erroneous. Hence, the estimates of evacuation times and of the feasibility of timely evacuation for certain areas are incorrect.

3.2-2 In formulating Contention 3.2, the Board intended that testimony should not be generic, but that it should focus on how the high population density in the vicinity of the Indian Point plants would exacerbate consequences resulting from the failure of human factors assumptions to hold in a radiological emergency. (Memorandum and Order Formulating Final Contentions on Commission Questions 3 and 4, February 7, 1983, pp. 4-7)

A. EMERGENCY RESPONSE IN GENERAL

3.2-3 Research on human response to diverse emergencies suggests that basic principles of response do not differ materially during different types of emergencies, and that although the lives of affected populations are disrupted, nevertheless speculative fears about the inability of populations to operate emergency organizations and carry out a response under adverse conditions

are unwarranted. Contentions to the contrary are unsupported by social science, psychological theory or empirical evidence. Observations at Three Mile Island show that there is no material difference between response to radiological and non-radiological emergencies. This is because the response is due to the stress caused by an emergency, rather than to the specific agent that creates the emergency. (Lecker Testimony, pp. 1, 3, 6; DuPont Testimony, pp. 6-9)

3.2-4 Adults respond to stress, and particularly to stress caused by an emergency, by looking to leaders for direction. This is especially so in the initial phases of a disaster. Studies have shown that when unstructured groups are put under stress, they evolve leaders spontaneously. The presence of pre-existing strong leadership and a plan of action increases the response. (Lecker Testimony, p. 4; Tr. 12049-12052)

3.2-5 It was suggested at the hearings that the public would not follow instructions and that emergency workers would abandon their duties in a radiological emergency, because there is no definite ending to such an emergency. (Erikson Testimony, pp. 3-4) However, the lack of a definite end is not relevant to evacuation behavior during an emergency, since even in an emergency

with a definite ending the evacuation would take place while danger still exists.

3.2-6 Suggestions were made at the hearings of the existence of "radiation response syndrome," allegedly a species of post-traumatic stress syndrome, which, it was claimed, might have an adverse effect on evacuation response. (See, Lifton Testimony, generally; PA Ex. 46, admitted, Tr. 12454; Tr. 12448-12449, 12470) The Board finds, however, that even if radiation response syndrome exists, it is unlikely to be a significant factor in emergency response, since its proponents acknowledge that it would manifest itself only months or years after exposure to radiation. (Lifton, Tr. 12429-12433, 12446-12449, 12461-12467, 12470, 12475-12477)

3.2-7 Various predictions of poor public response to a civil radiological emergency have been based principally on analogies to the atomic bombings of Hiroshima and Nagasaki. However, that experience was very different from a power plant accident, since it involved a wartime context with resulting immediate, concentrated death and destruction. (Lifton, Tr. 12429)

B. RESPONSE OF CHILDREN

3.2-8 Children react to emergencies better

than adults, because (a) they look confidently to adults to provide care and (b) they use optimistic anticipation to deal with stress caused by an emergency. The presence of a familiar parent surrogate, such as a teacher, will allay many fears of children who are being evacuated. There was no evidence offered that psychological trauma to children would be caused by evacuation during a radiological emergency. (Lecker Testimony, pp. 4-5; Solnit, Tr. 10479-10480)

C. TRADITIONAL EMERGENCY WORKERS

3.2-9 Traditional emergency workers, such as police, firemen and traffic control officers, will not desert their posts in the face of possible radioactive contamination. (FEMA Testimony, p. 25; McIntire, Tr. 1613-1616). Emergency workers in fact entered the TMI and Ginna plants after the accidents there. (McIntire, Tr. 2173-2176) These workers choose their professions because they seek, not fear, crisis mastery. They welcome the sense of filling a useful and appreciated role. The good response of these workers would not differ in radiological emergencies. This was borne out at TMI. (Lecker, Tr. 11992-11994, 1203412035) The problem usually is often that all workers come in, there are too many workers, not too few. (Id.)

3.2-10 Special contracts or advance compensation arrangements are unnecessary in order to ensure that emergency response personnel will respond when they are needed. (Dynes Testimony, p. 7; J. McGuire, Tr. 3738-3739; Northrup, Tr. 4245-4246) As discussed in Finding 3.2-9, emergency workers are not motivated by contractual commitments or the knowledge that they will be paid at overtime rates, but by the desire for crisis mastery and the knowledge that one can be helpful rather than being a victim.

3.2-11 A number of intervenor and interested state witnesses questioned the response of emergency workers during a radiological emergency. (Erikson Testimony, pp. 3, 5-7; Everhart Testimony, attachment p. 1; Tr. 4403; Bates Testimony, pp. 2-3; Guido Testimony, p. 3; Tr. 4942) However, there was unanimity of agreement among expert and non-expert witnesses at the hearings, and the Board so finds, that the level of response of emergency workers to emergencies in the past has been excellent. (FEMA Testimony, p. 25; McIntire, Tr. 1540-1541; Husar/McIntire, Tr. 1613-1616; McIntire, Tr. 2138, 2173-2176, 2212; Urbanik, Tr. 2138; Davidoff/Czech Supplemental Testimony, p. 8; Podwal, Tr. 11880; Lecker Testimony, pp. 6-9; Lecker, Tr. 11992-11993; Dynes, 12009-12010; Lecker, 12034-12035; Dynes Testimony, pp. 7-8; Tr. 12045-12046;

Schmer, Scalpi, Tr. 12108-12111; Guido, Tr. 4942-4943; Northrup, Tr. 4245-4246; J. McGuire, Tr. 3738; Brooker, Tr. 10228; Erikson, Tr. 9570-9572)

D. OTHER EMERGENCY WORKERS

3.2-12 Civil Defense workers and those who have volunteered to help during a radiological emergency are motivated by the same factors as traditional emergency workers, and can be expected to show the same good responses. (Dynes, Tr. 12009-12010) Other emergency workers, such as bus drivers, can also be expected to respond well. First, having a defined role during an emergency makes one feel useful and potent, rather than helpless. Second, the fact that a public emergency exists and that one's contributions to others is vitally needed has been found to enhance pro-social responses. (Lecker Testimony, p. 8; McIntire, Tr. 2212-2213) Continued improvement in training and public education will provide a higher confidence level to these workers regarding the safety of their families. (FEMA Testimony, p. 25)

E. RESPONSE OF THE PUBLIC

3.2-13 It was suggested that the populace might not follow assigned evacuation routes, and that

this could impede evacuation. (Erikson Testimony, p. 9) However, as discussed under Contention 3.3, the evacuation time estimates were constructed on the conservative assumption that evacuation traffic traveled only on assigned primary routes, and that if local roads were also used this would decrease evacuation times, because there would be less traffic on the primary routes. (See, Findings 3.3-18 and 3.3-21)

3.2-14 The public will not panic, and will follow instructions during a radiological emergency. An emergency produces a desire for order, instruction and leadership. This causes people to turn to leaders and existing plans. People in emergencies exhibit rational action and seem to accept the emergency without emotion, a phenomenon known as "isolation of affect." Attitudes change during emergencies, and people follow existing blueprints, even though they criticized them before the emergency occurred. Studies have shown that these effects occur, and that there is no panic during emergencies. (Lecker Testimony, pp. 9-11) These conclusions are supported by over thirty years of natural and man-made disaster research. Panic occurs only when persons are faced with a highly visible and immediate threat to survival within an enclosed area, and escape routes are cut off. Non-cooperative behavior during

evacuation is almost always isolated. (FEMA Testimony, p. 24; McIntire, Tr. 2496; Lecker, Dynes, Tr. 12056-12058; PA Ex. 1, admitted, Tr. 2591) There was no panic at Ginna, among either emergency workers or the populace. (McIntire, Tr. 2216)

3.2-15 FEMA explicitly considered the possible effects of panic and psychological stress when developing NUREG-0654. Based on over thirty years of research, FEMA concluded that these factors have a negligible effect on evacuation response. (PA Ex. 1, p. 1-1) The statements contained in PA Ex. 1 are FEMA's current position, and are true and accurate. (McIntire, Tr. 2239)

F. SHADOW EVACUATION

3.2-16 Spontaneous evacuation, or "shadow evacuation," to the extent it occurs before the issuance of a general evacuation order, will reduce evacuation times, not increase them. Any movement of population out of the EPZ before the evacuation order will mean that there are fewer people in the EPZ than modeled in the evacuation time estimates. (FEMA Testimony, pp. 25-26; McIntire, Tr. 2171-2172) The plans call for the setting up of access controls on evacuation routes to ensure free passage of traffic out of the EPZ. (FEMA Testimony, p. 26)

3.2-17 There was indeed a "shadow evacuation" in the area around TMI -- that is, many more persons evacuated than were advised to do so. However, this happens in every evacuation, and was exacerbated in the TMI area by unclear and uncertain information given to the populace, and by lack of emergency planning. (Dynes, Tr. 11980-11981) Moreover, shadow evacuation is an orderly and gradual process. An evacuation is not like a footrace where everyone starts at the same time. Instead, it has a gradual start, builds to a peak, and then tapers off. (Dynes, Tr. 11980-11981; Lecker, Dynes, Tr. 12025-12028) Any shadow evacuation around TMI was due in large part to the uncertainty of public information, the lack of emergency planning, the fact that there was never any advice not to evacuate, and the fact that the selective evacuation advisory came on a Friday, with indications that there would be no school on the following Monday. (Erikson, Tr. 9570, 9579-9598, 9618; Dynes, Tr. 11980-11981; Lecker, Dynes, 12025-12028) Moreover, as discussed under Contention 3.3, even if shadow evacuation occurs, evacuation times should not increase. (See, Findings 3.3-22 and 3.3-23) The Board therefore finds that any shadow evacuation should have no negative impact on public health or safety.

3.2-18 Attitude surveys about how people would respond to an emergency are not useful predictors of

actual emergency behavior, especially when the behavior is predicted to occur several years in the future. (Lecker, Dynes, Tr. 12037-12041)

3.2-19 FEMA has very little concern that under the appropriate circumstances the public cannot be educated to follow instructions. (McIntire, Tr. 2514)

G. RESPONSE OF PARENTS OF SCHOOL CHILDREN

3.2-20 It was predicted that parents would go to schools to try to pick up their children in a radiological emergency, rather than allowing them to be evacuated by bus, thereby overloading the roads around schools. (Erikson Testimony, p. 8) However, this testimony was not expert testimony. (Erikson, Tr. 9572-9573) As discussed under Contention 3.3, even if this phenomenon were to occur, it would not result in increased evacuation times and thus should have no effect on public health or safety. (Urbanik, Tr. 2041) In past mass evacuations, as many as a third of the evacuating families departed without reuniting and without causing disruption of the evacuation. Thus the gathering of families cannot be considered a necessary precondition to a successful mass evacuation. (McIntire, Tr. 2518-2519) The above one-third figure was for evacuations carried out without pre-existing

detailed plans. The knowledge of parents in the Indian Point EPZ that plans exist to evacuate their children and to speedily reunite families can only increase the percentage of families that would evacuate without reuniting.

3.2-21 Continued public education should raise the confidence level of parents, with the result that, if children are evacuated directly from schools, fewer parents will drive to schools to try to pick up their children. (FEMA Testimony, p. 26)

3.2-22 Some intervenor witnesses stated that the plans were inadequate because they provided no mechanism to reunite parents with children who had been evacuated from schools. In fact, each county plan contains such a mechanism. (FEMA Testimony, pp. 26-27) The Indian Point emergency planning pamphlet describes school evacuation, school reception centers and the provision of bus service to these centers. (FOE Ex. 1, p. 7; UCS/NYPIRG Ex. 14, p. 7)

H. INVISIBILITY OF THREAT

3.2-23 The invisibility of the threat from radiation will cause reduced, not enhanced, fear, and will increase, not reduce, the likelihood of an effective

emergency response. (Lecker Testimony, pp. 11-12)

3.2-24 Fear increases with the visibility of danger, not the invisibility. The literature shows that visual memories are the major part of post-traumatic nuerosis. (Lecker, Tr. 12016-12017)

3.2-25 The response to emergencies typically begins with a warning (e.g., that a hurricane is coming), rather than physical detection by evacuees of the danger. Thus the evacuees respond to most emergencies as they would respond to a radiological emergency -- by reacting to information that danger exists, rather than by actual perception of the danger. Thus the argument that radiation is unique because it is invisible is not relevant to gauging the effectiveness of evacuation. (Lecker, Tr. 12023-12024)

3.2-26 In any evacuation there is the fear of being overtaken by harm. Those fleeing because a dam is about to burst have good reason to fear that flood waters may overtake them, even though they can see nothing. This is analogous to fear of being overtaken by, or driving through, a radioactive plume. Visibility of the danger (e.g., approaching flood waters) can only make the fear worse. Therefore, in the absence of proof,

the Board rejects the illogical position that an invisible radiation threat is more frightening than a visible threat.

I. MISCELLANEOUS FACTORS

3.2-27 Any rise in public distrust of authority in recent years is irrelevant to evacuation response. In an emergency, people turn to existing authority regardless of preexisting scepticism. All the literature supports the conclusion that people react to stress by looking to leaders for information and guidance. (Lecker, Tr. 11984-11988)

3.2-28 UCS/NYPIRG witness Kai Erikson asserted that radiological disasters are not comparable to other disasters (Erikson Testimony, pp. 34), yet in his book on the Buffalo Creek disaster he states explicitly that all sorts of emergencies are similar, and that Buffalo Creek can serve as a good example of what will happen in the event of nuclear war. (Lecker, Tr. 11991, 12059-12061)

3.2-29 People are not afraid of radiation per se. Instead, they perceive greater or less danger depending on the source. Thus few people worry about the increase in dosage accumulated by taking a transcontinental airplane

flight, and they willingly submit to X-rays or radium treatments, whereas they are extremely fearful of exposure from a nuclear power plant. This difference has nothing to do with actual exposure levels. (Lecker, Tr. 11999-12000)

3.2-30 Those persons being evacuated are not helpless (as the intervenors seem to assume). Rather, research suggests that they can and do join in the rescue effort. (Dynes Testimony, pp. 7-8)

J. ULTIMATE FINDINGS

3.2-31 Past experience with mass evacuations shows that they are feasible and can be accomplished, and that people are helpful, not selfish, in providing assistance to others under evacuation conditions. On November 11, 1979, approximately 220,000 persons were evacuated from Mississauga, Ontario over a period of 24 hours, with no advance planning, and approximately 144,000 persons evacuated the area around Three Mile Island, despite confusing guidance from authorities. (Dynes Testimony, pp. 8-9)

3.2-32 Traditional emergency workers have always responded well to emergencies, and should do so in

a radiological emergency. Other emergency workers should also respond well. The public should cooperate in a radiological emergency and will not panic. Any shadow evacuation will not impede evacuation of the EPZ. The invisibility of the threat in a radiological emergency should not interfere with the response of emergency workers or the public. Therefore, the Board finds that there should be no significant difference between response to radiological and nonradiological emergencies. Consequently, the Board finds that assumptions made in the Indian Point emergency plans with respect to human response factors during a radiological emergency are not erroneous, and that the estimates of evacuation times and of the feasibility of timely evacuation for certain areas are not made incorrect by use of such assumptions. Therefore Contention 3.2 is denied.

IV. CONTENTION 3.3

3.3-1 Contention 3.3 reads as follows:

> The present estimates of evacuation times, based on NUREG-0654 and studies by CONSAD Research Corporation and by Parsons, Brinckerhoff, Quade & Douglas, Inc., are unreliable. They are based on unproven assumptions, utilize unverified methodologies, and do not reflect actual emergency plans.

3.3-2 Evacuation time estimates within the plume exposure emergency planning zone are addressed in Appendix 4 of NUREG-0654/FEMA-REP-1, rev. 1 (NUREG-0654).

A. BACKGROUND

3.3-3 The Radiological Emergency Response Plan (RERP) for each county presents the emergency preparedness, response and recovery activities to be enacted in that county in the event of an incident at Indian Point. The evacuation portion of each RERP, prepared by the licensees' consultant, Parsons Brinckerhoff Quade & Douglas, Inc. (Parsons Brinckerhoff), describes the scope of activities and the agencies that would be involved in an evacuation in response to an emergency at Indian Point. This information is contained in each RERP in Appendices A (general description of evacuation plans), G (population data for the plume and ingestion EPZs), H (list of special facilities with a population, temporary or permanent, of 15

or more persons) and K (ERPA boundaries) and Attachment 1 (evacuation procedures) of each agency's procedures. (Parsons Testimony, pp. 3-5)

3.3-4 Other supporting documentation prepared by Parsons Brinckerhoff includes the draft documents "Methodology to Estimate Roadway Travel Times During Evacuation," dated January 1981 (1/81 document); Methodology to Calculate Evacuation Travel Time Estimates for the Indian Point Emergency Planning Zone," dated November 1981 (11/81 document) (Con Edison Ex. 10, admitted, Tr. 11776); and "Guidelines for Updating the Evacuation Portions of the Radiological Emergency Plans and Procedures," dated April 1982 (4/82 document). (Parsons Testimony, pp. 3-4; Parsons Supplemental Testimony, pp. 4-6)

B. POPULATION ESTIMATES

3.3-5 The purpose of creating evacuation travel time estimates, as laid out in Appendix 4 of NUREG-0654, is to aid State and local officials in selection of an appropriate response option in the event of an incident at Indian Point. (Id., p. 19; Sears Testimony, pp. 29-30)

3.3-6 Parsons Brinckerhoff consulted extensively with county and state officials during the development and various revisions of the RERPs, which commenced in December 1979. (Parsons Testimony, pp. 6-8; Schmer/Scalpi, Tr. 12703-12709; J. McGuire, Tr. 3788; DelBello, Tr. 5868-5892; PA Ex. 20, admitted, Tr. 5870; 21, admitted, Tr. 5873; 22, admitted, Tr. 5876; 23, admitted, Tr. 5879; 26, admitted, Tr. 5891) Parsons Brinckerhoff also supplied the 4/82 document, designed to aid the counties in keeping their RERPs up to date. (Parsons Supplemental Testimony, p. 4) Although compensated by the Licensees, Parsons Brinckerhoff in effect acted as consultant for the State and counties (Parsons Testimony, p. 7)

3.3-7 The Indian Point plume exposure Emergency Planning Zone (EPZ) is broken down into 46 Emergency Response Planning Areas (ERPAs), based on a careful study of the EPZ. For purposes of preparing the estimates, persons to be evacuated were classified either as members of the residential population, special facilities population or transient population. (Parsons Testimony, p. 9) Residential population was identified using data consisting primarily of estimates of the 1980 population which were prepared by each of the four counties located in the EPZ. (Id.) The category of special facilities includes: schools (public/private elementary, middle and

high; nursery and day care); colleges; nursing homes, hospitals and other health care facilities; convents and monasteries; group homes; correctional facilities and police lockups; parks and camps. These facilities were inventoried through directories, letters and by telephone. (Parsons Supplemental Testimony, pp. 16-17) Parsons Brinckerhoff has supplied an updated list of special facilities. (Rosenblatt, Tr. 11916) The transient population includes people employed within the EPZ, people staying at hotels and motels in the EPZ, and visitors to parks and recreation areas. Employment estimates for the EPZ were based on data provided by the Tri-State Regional Planning Commission and by information obtained from telephone interviews with major employers. Hotels and motels in the EPZ were inventoried through state and local directories, and occupancy figures were obtained from telephone interviews. Data on park attendance for major parks and recreation areas in the EPZ were obtained from the Palisades Interstate Park Commission, the Taconic State Park Commission, and various park officials. (Parsons Testimony, p. 10) Estimates of the number of people to be evacuated and the way they are categorized are reasonable. (Urbanik Testimony, p. 6)

3.3-8 Population estimates in each category were adjusted for time-of-day and seasonal variations. (Parsons

Testimony, p. 12) The following six scenarios were considered for purposes of preparing evacuation time estimates, and different evacuation time estimates were developed for each scenario. (Id., pp. 15-16):

- night
- weekday, school in session
- weekday, school not in session
- evening
- weekend/holiday - summer daytime
- weekend/holiday - winter daytime.

3.3-9 The methodology used by Parsons Brinckerhoff for calculating the evacuation time estimates embodied in the Indian Point emergency plans has been reviewed and approved by the New York State Department of Transportation, the New York State Department of Health, the New York State Police and the New York State Radiological Emergency Preparedness Group (REPG). (Id., p. 35) It consists of four interrelated and frequently overlapping steps: (a) population identification and demand estimation, (b) modal splits, (c) roadway evacuation capacity analysis, and (d) traffic assignment. (Id., pp. 21-28)

3.3-10 Population identification and demand estimation involves estimating, for each of the six

scenarios, the numbers and locations of the resident general population (with and without privately owned vehicles available for evacuation), residents of special facilities, and transients. (Id., pp. 21-22)

3.3-11 Modal splits refer to the mix of travel modes by which individuals exit from the plume EPZ for each of the six scenarios. These are calculated based on the census, a 1980 survey by Yankelovich, Skelly and White and conversations with county officials and administrators of special facilities. If school is not in session, approximately 90% of the population is estimated to evacuate by car. If school is in session and school children are evacuated by bus, approximately 76% of the population is estimated to evacuate by car. The various estimates are set forth in Table A-3 of each County RERP (Id., pp. 22-24)

3.3-12 Roadway evacuation capacity analysis involves a survey of primary and backup evacuation routes. Every primary and backup route was traveled by Parsons Brinckerhoff personnel. (Id. supp. p. 7) The choice of evacuation routes was based on the existing roadway network. Data gathered for the routes included the number of lanes, lane width, shoulder width, location, timing of traffic controls, and speed limit. For routes to be followed by buses, information was also collected about

route length, location of existing transit stops, adequacy of stops, major concentration of potential users, street names and signing, tight turns, narrow or congested or one-way streets, and bridges with low clearances or weight limits. (Id., pp. 24-26; Parsons Supplemental Testimony, p. 3) FEMA is satisfied that buses can traverse their routes. (FEMA Testimony, pp. 27-28) Roads were examined outside of the EPZ, to the next major intersection, to ensure that no bottlenecks outside the EPZ could cause traffic to back up into the EPZ. (Della Rocca, Tr. 11944-11946) Capacities were calculated for each link in normal and adverse weather conditions (e.g., slipperiness from snow or ice and/or impaired visibility from fog or heavy rain). (Parsons Testimony, pp. 28-29) The estimated capacities at various locations were validated by comparison with actual volume figures from the New York State Department of Transportation. (Id., p. 25) The specific capacity calculation procedures are set forth in section III.A.1 and Appendix E of the 11/81 document. (Id., p. 26)

3.3-13 Parsons Brinckerhoff determined road capacities by use of the Highway Capacity Manual (Highway Research Board, Special Report 87, 1965). (Parsons Testimony, p. 21) It is the standard reference work for capacity of highways. (Urbanik, Tr. 1876-1878) Among the

road features considered in determining the capacity of each road segment were road width, shoulders, and impact of traffic signals. (Della Rocca, Tr. 11821)

3.3-14 The Highway Capacity Manual sets forth highway loadings (volume to capacity ratios) as "levels of service." Level of service A represents the least traffic and the highest speed, and level of service E represents the most traffic and the slowest speed. (Urbanik, Tr. 1889-1892) Level of service F represents forced feed, where a bottleneck exists. It is not continuous with level of service E, has no volume to capacity ratio, and is not useful for doing capacity analysis. (Urbanik, Tr. 1982-1893, 1900-1901)

3.3-15 Parsons Brinckerhoff assumed full-capacity use of evacuation routes (level of service E), but performed a conservative analysis by using level of service D (in which fewer cars per hour pass over a given segment), to account for possible organizational problems. (Urbanik, Tr. 1911-1912, 1949-1953, 1992-1995) A further capacity reduction of 20% was made to account for possible adverse weather conditions. (Urbanik, Tr. 1952-1956, 1992-1995) These adjustments are not required by NUREG-0654. Nevertheless, they introduce reasonable conservatism to account for organizational problems and weather conditions, and are recommended by the New York State Department of

Transportation, a noted authority. (Urbanik Testimony, p. 7; Tr. 2021-2022)

3.3-16 Traffic assignment involves calculation of travel times and delays due to the demand. This calculation is based on demand estimates, modal split and roadway evacuation capacity. The methodology is contained in the 11/81 document in section IV.D and Appendix G. A "static" model was used, which assumed instantaneous loading and concurrent vehicular demand on all roadway segments in the evacuation network. These results were compared with results calculated for test routes in Westchester and Rockland counties using a complex "dynamic" model being developed for the Federal Highway Administration. The results matched at a 97-98% aggregate correlation, thereby confirming the validity of the static model. (Parsons Testimony, pp. 26-28)

3.3-17 Evacuation routes were specifically designed to minimize delays from cross-traffic. (Della Rocca, Tr. 11853-11854, 11919)

3.3-18 The evacuation roadway network is reasonable because it considers the principal roadways that would be used in an evacuation. In actuality, some evacuees are likely to use other roadways not included in the evacuation roadway network. (Urbanik Testimony, p. 7)

This should reduce estimated evacuation times by reducing loads on primary routes. (Parsons Testimony, p. 34)

3.3-19 The accuracy of the licensees' analysis was independently verified by a FEMA consultant, who drove the roads in the EPZ and surrcunding area and performed independent calculations of volume-to-capacity ratios. This work showed that the licensees' analysis was reasonable. (Urbanik Testimony, p. 8)

C. ASSUMPTIONS MADE

3.3-20 The assumptions made in calculating evaluation times are contained in the 11/81 document. (Parsons Testimony, p. 31) A number of these assumptions were known to be conservative, and thus result in estimates that overstate the actual times required. These assumptions include (id., pp. 32-34):

- The evacuation travel time for all persons is presumed to be the time for the last person leaving the EPZ, even though significant portions of the population will have been evacuated well before the last person clears the EPZ boundary.

- Certain population groups (such as persons both residing and working in the EPZ) are double-counted.

- Zero absenteeism in schools is assumed.

- The weekend/holiday - summer daytime scenario assumes peak park attendance figures.

- Roadways were assumed to operate in their normal fashion, with no lane reversals. (See, Parsons Supplemental Testimony, p. 7) If drivers use the oncoming lanes, this could result in lowered evacuation times.(Podwal, Tr. 11898) In no case should this result in a lengthened evacuation time. (Podwal, Tr. 11900)

- It was assumed that persons without cars would not get rides from neighbors. A survey of the area showed that 74% of those with cars would offer a ride to a neighbor without a car. The licensees' booklet "Indian Point, Emergency Planning and You" encourages ridesharing. If half of those who said they would give a ride to neighbors did in fact give a ride to one person, the need for buses for the general public could be eliminated. (Parsons Supplemental Testimony, pp. 13-14) A survey introduced by intervenors indicated that only 3% of the Westchester County members of the general public who reside within the EPZ would require public transportation in the event of an evacuation (Seasonwein Testimony, p. T-17)

- For two-lane, two-way roads, 80% of the capacity was assigned to outbound traffic. It is anticipated that 90% of the capacity would be used for outbound traffic.

- Only 80% of buses and 50% of ambulances in the plume exposure EPZ are assumed to participate in the evacuation.

- All traffic was assumed to travel on the primary evacuation routes. To the extent that there is evacuation along local roads not identified as primary evacuation routes, evacuation time will be reduced, due to decreased loads on designated routes.

- All buses were assumed to carry 40 adults. The adult capacity of a school bus is 40-44, and the adult capacity of transit buses varies from 42-53. Standing room increases these capacities by roughly 50%.

3.3-21 To the extent that there is spontaneous evacuation within the EPZ (i.e., persons leave before a call for evacuation goes out), evacuation travel tires will be reduced. (Parsons Supplemental Testimony, p. 11)

3.3-22 If parents were to go to schools to pick up their children rather than allowing them to be evacuated by bus, total evacuation times will probably decrease, because there will be fewer buses on the roads. (Urbanik, Tr. 2041) Based on this reasoning, dismissal of schools at the alert stage (the so-called "O'Rourke plan" -- See, Findings 3.7-6 through 3.7-9) should decrease the total evacuation time for the school-in-session scenario, by

alleviating congestion around schools and by allowing buses to pick up all transit-dependent persons in a single wave, rather than evacuating schools first and then returning to the EPZ to pick up other transit-dependent persons.

3.3-23 Any spontaneous evacuation outside the EPZ should have little effect on evacuation times. (Parsons Supplemental Testimony, p. 12; Della Roca, Tr. 11863) The plans provide for traffic controls outside the EPZ, and also provide for limiting on-access to evacuation routes outside the EPZ. These traffic control measures, combined with the greater diversity of the road network outside the EPZ, would mitigate the effects of any spontaneous evacuation outside the EPZ. (Della Rocca, Tr. 11941, 11943-11945) As noted above, the evacuation time estimates specifically model traffic beyond the EPZ out to the next major intersection, to make sure that bottlenecks outside the EPZ will not result in backups into the EPZ. This modeling is based on actual traversal of roads outside the EPZ by Parsons Brinckerhoff representatives. (Della Rocca, Tr. 11944-11946) Road evaluations were personally confirmed by Staff witness Urbanik. (Tr. 2028-2030)

D. EVACUATION TIME ESTIMATES

3.3-24 Evacuation time estimates, broken down by ERPA, are presented in tabular form in Appendix A of each

county RERP and in the 11/81 document. (Parsons Testimony, p. 36) The estimates are presented for each of the six scenarios, for three sets of conditions: normal weather for both upper and lower travel times and adverse weather. (Id., pp. 28-29) The six scenarios and three weather conditions result in 18 evacuation time estimates for each ERPA. These estimates are further broken down into general population, special facilities population and transient population. Also given are estimates for 90° quadrants of the EPZ, at radii of two, five and ten miles from the plants. (Parsons Testimony, p. 31) The evacuation time estimates are also reflected in the licensees' own emergency plans, specifically PASNY Procedure IP-1017 and Con Edison Procedure IP-1013, which give a range of protective actions that may be recommended, based on the speed and other characteristics of any accident. (Sears Testimony, pp. 30-31)

3.3-25 A summary of these travel time estimates for each county follows (Hours:Minutes):

I. Westchester County

(a) Normal Weather Conditions

(i) Special Facility and Transient Population
Range of Evacuation Times: 0:50-12:40

(ii) General Population
Range of Evacuation Times: 0:35-10:40

(b) Adverse Weather Conditions

(i) Special Facility and Transient Population
Range: 0:50-15:25

(ii) General Population
Range: 0:40-12:55

II. Orange County

(a) Normal Weather Conditions
(i) Special Facility and Transient
Range: 1:40-12:35
(ii) General Population
Range: 0:40-12:20

(b) Adverse Weather Conditions
(i) Special Facility and Transient
Range: 1:50-15:30

(ii) General Population
Range: 1:35 - 15:15

III. Putnam County

(a) Normal Weather Conditions
(i) Special Facility and Transient
Range: 0:20-7:15

(ii) General Population
Range: 0:25-9:15

(b) Adverse Weather Conditions
(i) Special Facility and Transient
Range: 0:35-8:55

(ii) General Population
Range: 0:25-11:05

IV. Rockland County

(a) Normal Weather Conditions
(i) Special Facility and Transient
Range: 0:40-12:15

(ii) General Population
Range: 3:45-12:00

(b) Adverse Weather Conditions
(i) Special Facility and Transient
Range: 0:40-15:25

(ii) General Population
Range: 4:35-15:40

The above estimates include "terminal time" (the time required for cars to pass over local feeder streets to the first link of the designated primary evacuation route and for buses to make their pickups and enter the primary evacuation route), "roadway travel time" (the time required for all vehicles to traverse the primary evacuation route from the point of entry to the edge of the evacuated area) and "round trip time" (the time required for buses and other vehicles that must return to the EPZ to travel from the edge of the EPZ to the reception center, unload, return to the EPZ, make further pickups, and re-exit the EPZ). (Con Edison Ex. 10, admitted, Tr. 11776, pp. 41-43) The shorter (lower bound) evacuation times for normal weather conditions will result when (a) there are no unexpected long-run capacity restrictions on key links owing to incidents such as accidents, breakdowns and highway construction, (b) there is a high state of operational readiness (traffic control officers mobilized, traffic control devices operational, all buses stationed to begin their initial runs, etc.), and/or (c) an informed and cooperative public follows directions as instructed. The longer estimates for normal weather conditions will result when one or more of the above three conditions are not met. (Parsons Testimony, pp. 29-30)

3.3-26 It is expected that the evacuation times should be closer to the lower bound during normal weather conditions. In the evacuation before Hurricane Carla, there were fewer than the normal number of accidents and breakdowns. The relatively slow speeds likely in an evacuation in the Indian Point EPZ mean that any accidents that do occur are likely to be minor ones. (Podwal, Tr. 11939) The plans call for the stationing of towtrucks at strategic points, and the availability of these trucks has been confirmed. (Della Rocca, Tr. 11797)

3.3-27. A breakdown at a point on a highway where there are no shoulders can cause a blockage where the highway is operating at or near capacity. Such an event should have little effect during an evacuation, where the flow is almost all in one direction, since traffic can go into the left (oncoming) lane to go around the block. (Urbanik, Tr. 2031, Parsons Supplemental Testimony, p. 9; Della Rocca, Tr. 11793) Even assuming that accidents occur at double the normal rate, the effect on capacity is negligible. (Urbanik, Tr. 2032)

E. EVALUATIONS OF PARSON BRINCKERHOFF'S WORK

3.3-28 Parsons Brinckerhoff's evacuation time estimate methodology and implementation of that methodology

for Indian Point have received high praise from Brian K. Grimes, Program Director, NRC Office of Nuclear Reactor Regulation, and Darrel G. Eisenhut, Director, Division of Licensing, NRC Office of Nuclear Reactor Regulation. (Parsons Testimony, p. 35) An independent study performed for FEMA by CONSAD Research Corporation produced results that consistently matched the Parson Brinckerhoff estimates. What differences there were showed longer estimates for Parsons Brinckerhoff's work. (Id., p. 36; Sears Testimony, pp. 31-32)

3.3-29 Staff witness Dr. Thomas Urbanik, II testified that he had examined the initial version of Parsons Brinckerhoff's work for the Licensees and CONSAD's work for the Staff, and had rated them "excellent" in their conformance with the requirements of NUREG-0654/FEMA-REP-1, rev. 0. (Urbanik Testimony, p. 3) When NUREG-0654/FEMA-REP-1, rev. 1 was published, the studies were revised, and are in full compliance with the requirements of that document. (Id., pp. 3-4; Tr. 1967-1969; Sears, Tr. 12299-12301)

3.3-30 The methodology uses accepted and proven transportation planning techniques. (Urbanik Testimony, p. 6) It is consistent with NUREG-0654, uses the best available data, makes reasonable assumptions, and is therefore reasonable. (Id.)

3.3-31 The Licensees' evacuation time estimates are derived from and based wholly upon the presently existing roads within the EPZ. The roads are therefore adequate to evacuate persons from the EPZ within the estimated times, and such time estimates can appropriately be used by emergency response decision makers to determine the preferred emergency response in the event of an emergency at Indian Point. (Urbanik Testimony, pp. 9-10)

F. INTERVENOR CRITICISM

3.3-32 Criticisms of Parsons Brinckerhoff's use of the Highway Capacity Manual were given during the hearings, principally by Robert L. Morris. (Morris Testimony, p. 1) However, Morris had never performed an evacuation time estimate (Tr. 9727), did not examine Parsons Brinckerhoff's methodology (Tr. 9733-9735), was unfamiliar with NUREG-0654 and did not claim that Parsons Brinckerhoff's work failed to meet its standards. (Tr. 9757, 9760) Moreover, his criticism was based on a fundamental misunderstanding of the Highway Capacity Manual's level of service schema (compare Morris Testimony, pp. 1-2; Tr. 9739, 9802-9803, with Urbanik, Tr. 1889-1893, 1900-1901, 1949-1950) and used factual assumptions that were physically impossible. (Morris, Tr. 9750-9752)

3.3-33 A large number of intervenor witnesses asserted that their testimony addressed Contention 3.3. See, "Index of Witnesses Presenting Testimony on Emergency Planning Issues on Behalf of the Intervenors: UCS, NYPIRG, RCSE, WBCA, Parents, WESPAC, FOE, and Audubon, and on Behalf of the Attorney General of the State of New York," passim. However, none of this testimony asserted, in more than conclusory fashion, noncompliance with any planning standard or criterion of NUREG-0654, or took issue with the methodology or assumptions used in developing evacuation time estimates, or the plan provisions based on these estimates. Instead, these witnesses simply asserted that the roads in the area run primarily in a north/south direction, that many of them are narrow and winding, and that, in the opinion of the witnesses, the roads are insufficient to allow a timely evacuation. Such conclusory assertions are not useful in considering Contention 3.3.

G. ULTIMATE FINDINGS

3.3-34 The Board concludes that the evacuation time estimate portions of the county PERPs both satisfy the requirements of NUREG-0654 and provide the counties with an adequate framework for implementing an evacuation. The evacuation plan and procedures include the data necessary for emergency managers to make decisions concerning the

selection of protective actions as well as the operational strategies for carrying out evacuation-related activities. The plan has been developed to respond to a wide variety of situations while still maintaining an essential degree of flexibility.

3.3-35 The Board further finds that the evacuation travel time estimates contained in the RERPs and companion documents are reasonable representations of the time which can be expected to be actually required to evacuate some or all of the EPZ. The estimates are based on accepted transportation engineering methods applied in a regular manner, and are a realistic analysis of the area's existing transportation facilities and population subgroups, and of the functional responsibilities of county and local agencies. The RERPs will effectively accommodate future changes to any of these elements and, thereby, will remain up-to-date plans for responding to an incident at the Indian Point site.

3.3-36 The Board therefore finds that Contention 3.3 should not be adopted.

V. CONTENTION 3.4

3.4-1 Contention 3.4 reads as follows:

> The administrative control of notification procedures at Indian Point Units 2 and 3 is so deficient that the Licensees cannot be depended upon to notify the proper authorities of an emergency promptly and accurately enough to assure effective response.

3.4-2 Administrative controls of notification procedures at Indian Point Unit 2 have been completely revised and strengthened in recent years, and operating personnel have been retrained in these revised procedures (Jackson, Tr. 11753-11754) Reports are now made of incidents that have no potential for harm to the public, such as the dispatch of a person from the site to a local hospital for reasons unconnected with radiological matters. (Jackson, Tr. 11748-11750)

3.4-3 A dedicated hotline is now in place, linking the plants with the four counties and the state. Any earlier problems with confusion and lack of reporting by the Licensees have been cleared up by implementation of the new specific emergency classification system mandated by NUREG-0654. Since the effective date of these requirements, reporting from both Licensees has been acceptable, and has shown consistent improvement. (Davidoff/Czech Testimony, pp. 10-11)

3.4-4 The Licensees have the capability to notify offsite authorities of plant conditions, as required by 10 CFR, part 50, appendix E.IV.D.3 and NUREG-0654, Appendix 1. (Sears Testimony, pp. 49-50) Con Edison Procedure IP-1002 and Power Authority Procedure IP-1030 fulfill the requirements of the regulations and NUREG-0654, and, based on personal interviews with Licensee personnel, the NRC Staff found that these personnel understand the procedures and their responsibility for implementing them. (Id.) Moreover, the Licensees demonstrated prompt notification ability in the March 9, 1983 exercise. (Sears, Tr. 12240-12241, 14644) The Licensees have been conscientious about reporting incidents to the NRC. 10 CFR § 50.72 requires reporting within one hour. The Licensees always meet this requirement, and often report within five minutes. In fact, the Indian Point Personnel report more quickly and more completely than the personnel at any other plant in the country. (Sears, Tr. 12372-12377)

3.4-5 The Licensees have provided governmental authorities with means for obtaining key information independently. Each county in the EPZ has an Apple computer terminal, supplied by the licensees, which can be used to access the on-site MIDAS system, thus making information available at any time. (J. McGuire,

Tr. 3795-3797, 3880, 3888-3889; Schmer/Scalpi, Tr. 12115)

3.4-6 The Westchester County government is satisfied with current, improved levels of notification and procedures for notification. (DelBello, Tr. 5897-5899) Any earlier problems with reporting by the licensees to the Orange and Putnam County governments have now been cleared up, and there is now, if anything, over-reporting. (Schmer, Tr. 11213-11214; Scalpi, Tr. 11214) Communication between the operators of Indian Point and the Town of Yorktown has been greatly improved since 1981. (Eliot Testimony, p. 2)

3.4-7 Some intervenor testimony on Contention 3.4 cited an incident in October 1980 in which leaks from the fan cooler system caused a partial flooding of the containment at Indian Point Unit 2. (See, e.g., Ancona Testimony, pp. 3-4) However, administrative controls and procedures have been substantially upgraded since that incident, and largely as a result of it. (Jackson, Tr. 11748-11754)

3.4-8 Richard Altschuler testified that his survey showed that people in the area of Indian Point do not trust Con Edison to give accurate information and sound advice on what to do during a radiological emergency.

(Altschuler Testimony, p. 4) However, as FEMA pointed out, this alleged lack of confidence of the populace in Con Edison is irrelevant to response to an evacuation order, since that decision would be made by elected officials, who would be the source of the order. (McIntire, Tr. 2570) As we have noted in our Contention 3.2 findings (See, e. g. Findings 3.2-4 and 3.2-14), the public responds positively to directions from leaders. Moreover, public lack of confidence in Con Edison is irrelevant to Contention 3.4 as finally formulated, since it has nothing whatever to do with the administrative controls of notification procedures at Indian Point. Other witnesses stated a general lack of confidence in the Licensees (Co Testimony, p. 3; Gersony Testimony; Waterfall Testimony) or made vague assertions that lines of communication had not worked in the past. (Elliot Testimony, p. 2) This testimony is similarly irrelevant to Contention 3.4.

A. ULTIMATE FINDING

3.4-9 Administrative controls of notification and reporting procedures at Indian Point have been substantially upgraded, to the point that the NRC Staff finds them exemplary. State, county and local governments expressed approval of current reporting. The Board therefore finds that administrative controls and procedures

are sufficient to give reasonable assurance that the licensees can be depended upon to notify the proper authorities of an emergency promptly and accurately enough to assure effective response. Therefore, Contention 3.4 is denied.

VI. CONTENTION 3.6

3.6-1 Contention 3.6 reads as follows:

> The emergency plans and proposed protective actions do not adequately take into account the full range of meteorological conditions for Indian Point Units 2 and 3.

3.6-2 The emergency plans and proposed protective actions for Indian Point take into account both good and adverse weather conditions. (Schwartz Testimony, pp. 2-3)

3.6-3 The evacuation time estimates developed by Parsons Brinckerhoff account for varying meteorological conditions. (See, Findings 3.3-16, 3.3-24, 3.3-26) In particular, if roads are blocked, general sheltering may be the preferred protective response. (See, e.g. Westchester RERP, p. III-50)

3.6-4 Inversion conditions will decrease dispersion of airborne radioactive materials. However, dispersion estimates for atmospheric conditions including inversions are included in the dose projection capability. (FEMA Testimony, p. 32.)

3.6-5 Snow and icing conditions would increase the time necessary for emergency organizations to respond

and evacuations to take place. However, these increased times are reflected in the evacuation time studies. (FEMA testimony, p. 33.)

3.6-6 If a severe snowstorm (12-16 inches) were to occur at the worst time (4:00 p.m.), with wet, heavy snow and temperatures in the most difficult range (high 20s to low 30s), the main roads in Westchester could be made thoroughly clear in 4-6 hours. (Bohlander, Tr. 5764-5765, 5768) Roads could be made passable in 4-6 hours after an ice storm, by application of sand. (Bohlander, Tr. 5767)

3.6-7 Any release occurring during rain or snow would be quickly scavenged and pulled down very close to the plant. See, Finding 4.2(c)-8) Any release during cold weather would cause the plume to form a cloud, causing self-rain-out that would also pull the plume down very close to the plant. See, Finding 4.2(c)-8)

A. ULTIMATE FINDING

3.6-8 Meteorological conditions have been appropriately taken into account in the current plans. There was no evidence of unique meteorological conditions in the

vicinity of Indian Point that could affect emergency planning in a way which is different from the consideration of meteorological factors at other nuclear power plants. (Davidoff/Czech Testimony, p. 12) The Board therefore finds that no case has been made for adoption of Contention 3.6 and accordingly the Contention is denied.

VII. CONTENTION 3.7

3.7-1 Contention 3.7 reads as follows:

> The problems of evacuating children from threatened areas have not been adequately addressed in the present emergency plans.

3.7-2 Since children who are with their parents would evacuate with them, problems arising during such an evacuation would not be problems peculiar to the evacuation of children. Thus Contention 3.7 deals only with situations in which children are not with their parents at the time an evacuation order is given. The situation that generated most of the testimony under Contention 3.7 concerned evacuation while school is in session.

A. EVACUATION FROM SCHOOLS

3.7-3 The problems of evacuating children are specifically addressed in the RERPs. Entire sections of the RERPs are devoted to evacuating the schools and special facilities which host children. (See, e.g. Westchester County RERP, Procedure 4, Putnam County RERP, Procedure 4) Parsons Brinckerhoff originally designed a two-wave evacuation whereby schools would be evacuated first, and the children taken by bus to school reception

centers outside the EPZ, where they would be picked up by their parents. The buses would then return to the EPZ to pick up other transit-dependents. (Parsons Testimony, pp. 17-18, Parsons Supplemental Testimony, pp. 14-16; Orange County RERP, § III.H.5, Appendix E; Westchester County RERP § III.H.5, Appendix C; Rockland County RERP, § III.H.6(I); Putnam County RERP, § III.H.5, Appendix A) This plan was found acceptable by the State of New York. (Davidoff/Czech Testimony, p. 13; Davidoff/Czech Supplemental Testimony p. 13) The two-wave evacuation plan for school children was successfully tested in Putnam County in the March 3, 1982 exercise by actually carrying out a simulated evacuation of one group of school children. (Scalpi, Tr. 12121)

3.7-4 The principal emphasis of intervenor testimony under Contention 3.7 was that parents would not trust evacuation of their children from schools by bus, and would insist on picking their children up at school. (Capon Testimony, p. 2; Churchill Testimony; Courtney Testimony, pp. 1-2; Doughty Testimony; Ford Testimony, p. 2; Gochman Testimony; Gunn Testimony, p. 2; Helbraun Testimony, p. 2; A. Johnson Testimony, p. 1; Kahn Testimony, p. 1; Lavelle Testimony, pp. 1-2; Puglisi Testimony, p. 2; D. Smith Testimony, p. 9; Ziegler Testimony, p. 2)

3.7-5 Although one of the objections of intervenors was that parents would not trust school evacuation of their children, certain intervenor and county actions can only be viewed as tending to foster this mistrust. The survey distributed by the Schools Subcommittee of the Rockland Citizens Advisory Committee (Con Edison Ex. 9) was viewed by School Superintendent Leonard Scharf, a member of the Subcommittee, as "misleading" and "inflammatory." (Tr. 11210-11211) The Board agrees with Mr. Scharf. Similarly, a communication to school principals from Parents Concerned About Indian Point (PA Ex. 44) mischaracterized the sorts of services which schools would be expected to provide in a radiological emergency and used inflammatory language. Mr. Scalpi, Putnam County's Civil Defense Director, testified that the survey had a "devastating" effect on the County's efforts to plan for that County's schools (Tr. 12089-12090)

3.7-6 The intervenors' attack on the school-in-session scenario was made obsolete by the proposal by Westchester County Executive Andrew P. O'Rourke that schools be dismissed and students be sent home at an early stage of any radiological emergency, to evacuate with their families or others. (O'Rourke Testimony, pp. 2-3, Tr. 11525-11526; Davidoff/Czech Supplemental Testimony, pp. 5, 9; Davidoff, Tr. 11386; Schmer/

Scalpi, Tr. 11217-11218)

3.7-7 The option to dismiss schools early, using generic snow-day procedures required by state law, has always been available under the plans. (Scalpi, Tr. 12118) Nor is it a new or unfamiliar concept. It has been used some 20 times in Orange County in the last five years, and has proved workable. (Schmer, Tr. 12118-12120) An advantage of early dismissal is that it allows school bus drivers to drive their regular routes. (Podwal/Della Rocca, Tr. 11906-11907)

3.7-8 The early school dismissal plan was tested in the March 9, 1983 exercise and found to be workable. (FEMA Post Exercise Assessment, April 14, 1983, p. 29; Davidoff/Czech Supplemental Testimony, p. 9; O'Rourke, Tr. 12525-12526; Schmer/Scalpi, Tr. 12120-12121)

3.7-9 The New York Education Law requires each school district, and parents, to have pre-existing arrangements for go-home situations. Each family is required to have on file with the school district instructions as to what to do when school is dismissed early and there is no adult at home. (Davidoff, Tr. 11464; McIntire, Tr. 10526-15027) Thus the O'Rourke plan does not require schools to do anything other than what they

are already prepared to do.

3.7-10 The emergency planning brochures mailed to residents in the EPZ specifically address the need for planning for "latch key" children who may be at home without an adult present. The tearout card in the brochure has a place for parents to indicate that children may be alone and could require assistance in a radiological emergency. (FEMA Testimony, pp. 15-16) FEMA has approved plans in New York and New Jersey for various non-radiological emergencies. In none of these plans has FEMA required that special provisions be made for latch-key children. (McIntire, Tr. 2302-2303)

3.7-11 Experience in other disasters has shown that individuals rely on the extended family and other important social contacts during warning and response periods in disasters, and that friends and relatives will assist special populations, such as latch-key children, in an emergency situation. (FEMA Testimony, p. 16)

3.7-12 Meetings between the New York State Radiological Emergency Preparedness Group and school officials in the four counties have continued on an ongoing basis to solve problems as they arise. There has been a marked increase in school officals' familiarity with the

plans and their implications to schools. The New York Radiological Emergency Preparedness Group places a high priority on this process. (Davidoff/Czech Supplemental Testimony, p. 9)

3.7-13 During the March 9, 1983 exercise the State of New York offered to use National Guard forces to operate buses in Westchester County, but County Executive O'Rourke replied that since the schools had been dismissed early, there were enough buses and drivers, and the National Guard was not needed. (Davidoff, Tr. 11479-11480)

B. OTHER CONSIDERATIONS RELATING TO EVACUATION OF CHILDREN

3.7-14 Other intervenor criticism of plans for evacuation of children fell into five basic areas: (a) reuniting of families (b) increased evacuation times, (c) trauma to children, (d) panic of children and parents, and (e) availability of buses.

3.7-15 Many of the witnesses asserted that no provision had been made in the plans to reunite families that had not evacuated as a unit. However, the plans do in fact contain provisions to reunite families. (See, Finding 3.2-22)

3.7-16 Many witnesses stated that if parents go to schools, roads in the vicinity of schools will become jammed, thus impeding evacuation. However, the uncontroverted expert testimony before the Board is that this will not increase evacuation imes, and may even decrease them. (Urbanik, Tr. 2041)

3.7-17 A number of witnesses stated that evacuation of children separate from their families would result in severe trauma to such children. As discussed under Contention 3.2, children react well to emergencies, because they turn to adults for aid, particularly familiar parent surrogates such as teachers, and because they use optimistic anticipation to deal with stress. (See, Findings 3.2-18)

3.7-18 A number of witnesses predicted that children and parents would panic if it became clear to them that they would not evacuate as a family unit. As shown under Contention 3.2, however, actual panic occurs only under very special circumstances that would not be present during an evacuation in the area around Indian Point. Panic did not occur at Three Mile Island, despite the fact that a substantial percentage of the evacuating family units were not complete. (See, Findings 3.2-14, 3.2-15 and 3.2-20)

3.7-19 Appendix A and Procedure 4 of each county RERP provide information regarding the identification and location of each school within the EPZ, enrollment size, number of buses required for evacuation of the population of each school, and the name of the servicing bus company. The plans provide for bus service for all transit-dependent persons. (FEMA Testimony, p. 33) There are enough buses available to ensure evacuation of children in all counties of the EPZ. (Davidoff, Tr. 11439-11440; Schmer, Scalpi, Tr. 12117)

C. ULTIMATE FINDING

3.7-20 Implementation of the O'Rourke plan makes objections to the school-in-session scenario obsolete. The O'Rourke plan produces no insoluble problems with latch-key children. If parents come to schools, evacuation times will not be increased. Trauma to children from evacuation from school will be minor, and children and parents will not panic. An ample supply of buses is available. Therefore, the Board finds that Contention 3.7 should not be adopted.

VIII. CONTENTION 3.9

3.9-1 Contention 3.9 reads as follows:

> The road system in the vicinity of the Indian Point plant is inadequate for timely evacuation.

3.9-2 Neither the NRC's regulations nor NUREG-0654 specifies any particular time period for evacuation. Therefore, under Commission Question 3, "timely evacuation" is not a requirement.

3.9-3 As discussed at length under Contention 3.3, the radiological emergency response plans and the evacuation time estimates incorporated in those plans were prepared based on existing roads, using thoroughly validated methodologies and conservative assumptions about traffic loads during an evacuation.

3.9-4 The evacuation time estimates were further validated by an independent study conducted for FEMA, which produced similar results. (See, Finding 3.3-28)

3.9-5 Raymond S. Jurkowski, Deputy Commissioner of the Westchester Department of Transportation, stated that his department had examined the methodologies used by Parsons Brinckerhoff, and that he had no criticism of the highway element of the plan. (Tr. 5293-5294) Such a

statement, from an official responsible for transportation matters, and based on actual examination and evaluation of the plans, and coming from the mouth of a party adverse to the licensees, is highly probative of the accuracy of Parsons Brinckerhoff's evacuation time estimates and the ability of the roads to carry the estimated traffic.

3.9-6 As discussed under Commission Question 1, by far the most probable accident scenario at Indian Point is a slowly evolving gradual overpressurization of the containment by steam pressure (Release Category 2RW) See, Finding 1.1-260. In this sequence, containment failure would occur no sooner than 12 hours, and more likely more than 24 hours, after the accident began. (Finding 1.1-149) The slow development of this sequence would allow ample time to terminate the accident and to mobilize to implement evacuation. Indeed, most people could walk out of the EPZ if necessary. (Findings 1.1-263, 1.1-362) While the 2RW scenario may entail moving large populations substantial distances, there is ample time to do so.

3.9-7 The only other dominant accident scenario at Indian Point is the interfacing system LOCA, or V sequence. (Finding 1.1-260) This sequence evolves quickly, often resulting in a release of radioactive materials in a

matter of a few hours. (Finding 1.1-260) It would therefore be impossible, even with perfect organization and roads, to evacuate before plume passage. An alternative response is therefore sheltering until the plume has passed, followed by prompt relocation of those persons in highly contaminated areas. (Findings 1.1-265 to 1.1-268) There is no significant difference in consequences in the V sequence between immediate evacuation and sheltering followed by early relocation. (Finding 1.1-268) Since only contaminated areas would be evacuated after a V sequence accident, it would not be necessary to evacuate the entire EPZ. Thus evacuation efforts would be facilitated.

3.9-8 In a V sequence accident, the required response would be minimal. In a 2RW sequence acccident, there would be enough time to respond, even if help from the State is needed, and enough time to carry out an evacuation on existing roads before any release occurred. Thus, in neither case does individual risk depend on calculated evacuation times. (Findings 1.1-265 to 1.1-270, 1.1-263, 1.1-362, 5.0-38)

A. ULTIMATE FINDING

3.9-9 There is no requirement under the regulations for "timely evacuation." The evacuation time

estimates have been prepared based on existing roads, using valid methodologies and conservative assumptions. During a fast-moving accident the preferred response is sheltering followed by early relocation. During a slowmoving accident there would be ample time to evacuate. Therefore, the Board finds that no case has been made for adoption of Contention 3.9 and accordingly it is denied.

IX. CONTENTION 3.10

3.10-1 Contention 3.10 reads as follows:

> The emergency plan fails to conform to NUREG-0654 in that, contrary to Evaluation Criterion II.J.10.d, proper means for protecting persons whose mobility may be impaired have not been developed. Specifically, adequate provisions have not been made for groups named in the bases submitted for the following contentions:
>
> WESPAC 6
> Parents I, basis (22) and II, basis (7)
> UCS/NYPIRG I(B)(2), basis (6) and I(A), basis (7)

3.10-2 The groups named in the above bases are as follows: WESPAC 6 names (a) patients at the Franklin Delano Roosevelt Hospital in Montrose, (b) senior citizens, (c) handicapped children at the Asthmatic Children's Foundation Home in Ossining and the Blythedale Children's Hospital in Valhalla,* (d) residents at the New York School for the Deaf in White Plains,* (e) inmates of the Ossining Correctional Facility, (f) residents of community residences of the Westchester Association for Retarded Citizens, (g) residents of nursing homes such as the Skyview Nursing Home in Croton-on-Hudson, and (h) large numbers of people, particularly young persons, who may be expected to be at parks and outdoor areas during the summer. Parents I, basis 22, names "many special institutions" within the

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* Note: this facility is beyond the approximately 10-mile plume exposure EPZ (See, CE Ex. 8)

EPZ, including the Asthmatic Children's Foundation. Parents II, basis 7, names special institutions outside the EPZ, such as Blythedale Children's Hospital and the New York School for the Deaf, which may have their resources strained during an emergency. UCS/NYPIRG (B)(2), basis 6, names handicapped persons and invalids. UCS/NYPIRG I (A), basis 7, names the blind, the deaf, members of the population with learning disabilities, latch-key children and other special populations.

3.10-3 Each county plan contains a Special Facilities Procedures section that addresses the evacuation of individuals from special facilities. Evacuation steps for each type of facility are described. A table displays the number to locate each special facility shown on the map, the ERPA number, the population of the facility, a description of vehicles owned by the facility, a description of the number and type of vehicles needed for augmentation, and the telephone number of the facility. (FEMA Testimony, p. 41; Parsons Testimony, pp. 23-24) All hospitals and nursing homes in the EPZ are required to have evacuation plans, and it is doubtful that any such institution has failed to make such plans. (Kaminski, Tr. 5621-5622)

3.10-4 The plans also provide a sheltering option for persons for whom evacuation may not be possible,

such as hospital patients. (FEMA Testimony, p. 42)

3.10-5 The plans do not contain provisions for implementing protective actions at special facilities outside the 10-mile EPZ because this is not required by NUREG-0654. (FEMA Testimony, p. 42)

3.10-6 In "Draft Guidelines for Updating the Evacuation Portions of the Radiological Emergency Response Plans and Procedures," dated April 1982, a mechanism is provided to add any nursing home or other residence of mobility-impaired individuals, and to update lists of non-institutionalized mobility-impaired individuals. (Parsons Testimony, p. 14; Parsons Supplemental Testimony, p. 5)

3.10-7 The RERPs provide for buses, ambulances and other vehicles to evacuate the mobility impaired transit dependent population. (See, e.g. Westchester RERP, Section III. A.3 and III. B.3 and Attachment 3 of Appendix A)

3.10-8 Potassium iodide has been provided for mobility-impaired persons who cannot be moved. (See, Finding 4.2(a)-7)

3.10-9 Element J.10.d of NUREG-0654 does not specify a single method for protecting mobility-impaired

individuals. In many cases, sheltering may be the preferred protective response. Significant improvements in planning have or are scheduled to be undertaken for those institutions within the 10-mile EPZ that house mobility-impaired individuals. (FEMA Supplemental Testimony, p. 3)

3.10-10 The majority of special facilities own vehicles that would be used if evacuation becomes necessary. The plan identifies private bus companies that would be used to provide supplemental buses and vans. (FEMA Supplemental Testimony, p. 3) Of course, not all wheelchair-bound persons would be evacuated in wheelchairs. Many could sit in bus seats. (Della Rocca/Rosenblatt, Tr. 11800-11803) In Westchester County alone there are 105 county buses that have wheelchair lifts. (Jurkowski, Tr. 5325)

3.10-11 The revised county plans provide two primary methods for identifying the non-institutionalized mobility-impaired transit-dependent population: (a) the public information pamphlets distributed annually include a mail-in postal card, so that this population can be identified in advance, and (b) at the time of an accident, a special telephone number or numbers will be announced over the broadcast media for mobility-impaired persons to call to request transportation. (FEMA Supplemental

Testimony, p. 3)

3.10-12 Each of the four counties in the EPZ has agreed to contact appropriate agencies and facilities to solicit information about additional mobility-impaired persons. The agencies are the Westchester County Department of Social Services, the Orange County Department of Social Services, the Putnam County Fire and Rescue Service, and the Rockland Civil Defense Director. Each county will maintain and update a confidential list of non-institutionalized mobility-impaired persons. In case of a radiological emergency, each county EOC would be manned by a representative from the county agency responsible for the special problems of mobility-impaired persons. If evacuation becomes a possibility, the staff will activate its procedures, including notification to and coordination ith EOC command. (FEMA Supplemental Testimony, p. 3)

3.10-13 The policy of the New York State Department of Corrections is that inmates at the Ossining Correctional Facility will not be evacuated, but will instead be sheltered and provided with potassium iodide if necessary. The procedures of the Department of Corrections provide for staffing of the facility. (FEMA Supplemental Testimony, p. 4)

3.10-14 FEMA believes that most of the non-institutionalized mobility-impared transit-dependent persons within the EPZ live with other individuals or have neighbors who would assist in protecting them in the event of an accident. (FEMA Supplemental Testimony, p. 4)

3.10-15 It also seems logical to conclude that any mobility-impaired transit-dependent persons who do not live within the EPZ but who spend considerable time there would seek information from others, and in the event of an accident would listen to the Emergency Broadcast System. (FEMA Supplemental Testimony, p. 4)

3.10-16 Major responsibility for evacuation of mobility-impaired persons lies with the families and friends of those persons, in a radiological emergency just as in any other emergency. (Davidoff/Czech Testimony, p. 18) Arrangements have been made to evacuate all those who returned mail-in cards from the brochures, and new mailings of the brochures are being prepared. (Davidoff/ Czech Supplemental Testimony, p. 11)

3.10-17 In Orange County additional efforts to identify mobility-impaired persons have been made, including contacts with the police, utility companies (to

identify those on life support systems), ambulance corps, as well as personal contacts by telephone with the affected persons themselves. (Schmer, Tr. 12124-12125) Newspaper ads have been placed in both Orange and Putnam Counties. (Schmer/Scalpi, Tr. 12130-12131) Officials in Orange and Putnam state that they have enough buses and drivers to evacuate all mobility-impaired persons in their part of the EPZ. (Schmer/Scalpi, Tr. 12134-12135)

A. ULTIMATE FINDING

3.10-18 The intervenor witnesses for the most part simply asserted that the plans were inadequate to assure the evacuation of mobility-impaired transit-dependent persons. The few references to the specific institutions or groups mentioned in the contentions merely stated that these institutions existed and stated that in the opinion of the witness these institutions could not be evacuated. Such statements are insufficient in the face of testimony of FEMA, the State of New York the NRC Staff and the licensees' experts that the plans provide adequately for evacuation of such persons. The Board therefore finds that Contention 3.10 should not be adopted.

X. COMMISSION QUESTION 3

3.11-1 Commission Question 3 consists of two parts: the first part asks what is the degree and status of conformance of Indian Point state and local emergency planning in the Indian Point area with NRC and FEMA guidelines; the second part seeks to determine what the minimum hours warning for an effective evacuation of a 10 mile quadrant at Indian Point would be.

A. ULTIMATE FINDINGS

3.11-2 We have in our Findings under Contention 3.1 examined the question of compliance of emergency planning at Indian Point with the sixteen planning standards of 10 CFR § 50.47(b). We found that the plans for the Indian Point area are in compliance with these standards. In our findings under the other Question 3 contentions we examined claims that the Indian Point plans are deficient in particular areas and denied each of these claims. Based upon the findings we have made under the Question 3 contentions the Board finds that state and local emergency planning in the Indian Point area is in compliance with applicable federal regulations and that there is reasonable assurance that adequate protective measures can and will be taken in the event of radiological emergency. However, based upon FEMA's December 16, 1982

Update Report and its 1983 Post Exercise Assessment there are a number of deviations from particular NUREG-0654 evaluation criteria.

3.11-3 With regard to the second part of Question 3 we interpret it as seeking the range of times for an effective evacuation of quadrants within a ten mile radius of Indian Point depending upon the level of preparation or mobilization. The Parsons Brinckerhoff document "Methodology to Calculate Evacuation Travel Time Estimates for the Indian Point Emergency Planning Zone" (Con Edison Ex. 8) provides estimates of evacuation time for different sectors of the EPZ including estimates for 90° quadrants for a ten-mile radius The estimates for the four quadrants are as follows:*

I. Normal Weather Conditions

A. 10-Mile Radius:

(i) School in Session Scenario

	Resident Population		Special Facilities	Transients
	With Autos	Without Autos		
Sector**	From - To	From - To	From -To	From - To
I	6:06 -10:15	6:50 - 10:15	8:20 - 12:40	6:05 - 10:15
J	5:10 - 8:15	7:40 - 10:40	7:15 - 9:45	5:10 - 8:15
K	6:55 -11:40	7:15 - 12:00	7:55 - 12:15	6:55 - 11:40
L	5:30 - 9:25	6:00 - 9:50	5:50 - 9:40	5:45 - 9:40

* These figures do not include the 15 minutes allowed for licensee's notification of offsite officials, and the 15 minutes within which these officials are expected to notify the public.

** Sectors I, J, K and L are for approximately 90° quadrants of the full Indian Point plume exposure EPZ (Con Edison Ex. 10, p.5). Maps showing the areas included within each sector are shown in Figures 11 to 14 of Con Edison Ex. 10.

(ii) Nighttime Scenario

Sector	Resident Population With Autos From - To	Resident Population Without Autos From - To	Special Facilities From -To	Transients From - To
I	3:30 - 5:45	3:30 - 5:45	3:45 - 6:00	3:30 - 5:45
J	2:40 - 4:00	3:00 - 4:10	2:50 -4:10	2:40 - 3:55
K	3:40 - 6:10	3:50 - 6:15	3:55 - 6:25	3:20 - 5:35
L	2:40 - 4:25	3:05 - 4:55	3:00 - 4:45	2:40 - 4:25

II. Adverse Weather Conditions

A. 10-Mile Radius

(i) School in Session Scenario

Sector	Resident Population With Autos	Resident Population Without Autos	Special Facilities	Transients
I	12:40	12:40	15:25	12:40
J	10:15	12:55	11:55	10:15
K	14:30	14:50	15:10	14:30
L	11:40	12:05	11:55	11:50

(ii) Nighttime Scenario

Sector	Resident Population With Autos	Resident Population Without Autos	Special Facilities	Transients
I	7:05	7:05	7:20	7:05
J	4:50	5:05	5:05	4:50
K	7:35	7:45	7:50	6:55
L	5:25	5:55	5:45	5:25

Con Edison Ex. 10 at 46-49

3.11-4 Con Edison Ex. 10, the source for the above estimates, is one of the documents supporting Parsons Brinckerhoff's evacuation time estimates in the RERPs (See, Finding 3.3-4). In our findings under Contention 3.3 we found the Parsons Brinckerhoff estimates in the RERPs to be reasonable, in full compliance with NUREG-0654, and to provide an adequate framework for evacuation. (See, Findings 3.3-34 and 3.3-35) Accordingly, the Board finds that the estimates set forth in Finding 3.11-3 to be reasonable.

FINDINGS OF FACT
UNDER COMMISSION QUESTION 4

I INTRODUCTION

4.0-1 Commission Question 4 reads as follows:

> What improvements in the level of emergency planning can be expected in the near future, and on what time schedule, and are there other specific offsite emergency procedures that are feasible and should be taken to protect the public?

4.0-2 In our February 7, 1983 Memorandum and Order we admitted three contentions under this Question, 4.1, 4.2 and 4.7. Contention 4.2 has four subparts (a) through (d). The Board will first address the first part of Question 4 which seeks to determine the improvements in the level of emergency planning which could be expected in the near future. We will then address the Contentions which were admitted under the second part of Question 4 as proposed specific offsite emergency procedures.

4.0-3 In the over two years since the Commission proposed Question 4 in its January 8, 1981 Order there have been steady and marked improvements in virtually all facets of emergency planning. These improvements are obvious from a review of the various FEMA and New York State REPG documents. We have discussed many of their changes in our

findings under Contention 3.1. In addition to the improvements since January 8, 1981 further improvements in training and resources are expected. In addition to these planned changes, we have in our Recommendations suggested a number of other proposed changes in emergency planning.

II. CONTENTION 4.1

4.1-1. Contention 4.1 states:

> The plume exposure pathway EPZ should be expanded from its present 10-mile radius in order to meet local emergency needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries.

A. BASIC REQUIREMENTS

4.1-2 NRC regulations (10 CFR §§ 50.47(c)(2), 50.33(g)), NUREG-0654 (pp. 10-12), and the Orders of the NRC in this proceeding (Consolidated Edison Company of New York (Indian Point, Unit 2), Power Authority of the State of New York, (Indian Point, Unit 3), CLI-82-15, 16 NRC 27, 36 (1982)) all establish that the plume exposure pathway EPZ is to have a radius of "about ten miles." In the recent case of Southern California Edison Company (San Onofre Nuclear Generating Station, Units 2 & 3), LBP-82-39, 15 NRC 1163 (1982), an Atomic Safety and Licensing Board conducted an extensive review of the development of the generic requirement of an EPZ radius of about 10 miles. That Board concluded that the language of the regulation is mandatory, that the generic requirement is based on extensive research, set forth in NUREG-0396, and that any deviations from a 10 mile radius must be minor ones. (15 NRC 1163, 1178-1182)

4.1-3 The basis for an EPZ of about 10 miles was developed by a joint NRC/EPA task force, which produced results reported in NUREG-0396/EPA 520/1-78-06, "Planning Bases for the Development of State and Local Government Emergency Response Plans in Support of Light Water Nuclear Reactors" (NUREG-0396). (Schwartz Testimony, p. 3) The material presented in NUREG-0396 was developed to cover the broad spectrum of accidents, rather than only severe accidents, in order to cover all relevant points. (Id., p. 4; NUREG-0396, p. 6) The study considered all WASH-1400 scenarios. (Schwartz Testimony, p. 4) NUREG-0396 was issued before the accident at Three Mile Island. However, comments received in the wake of TMI reinforced the initial determination on the proper size of the EPZ. (Id.) NUREG-0396 criteria were adopted in NUREG-0654 at p. 10.

4.1-4 The size (about ten miles) of the plume exposure EPZ was based on the following considerations:

a. projected doses from the traditional design basis accidents would not exceed Protective Action Guide levels outside the zone;
b. projected doses from most core melt sequences would not exceed Protective Action Guide levels outside the zone;
c. for the worst core melt sequences, immediate life threatening doses would generally not

occur outside the zone;

d. detailed planning within 10 miles would provide a substantial base for expansion of response efforts in the event that this proved necessary.

The NUREG-0396 report also illustrates the relative effectiveness of shelter versus evacuation at various distances, and indicates that shelter with subsequent relocation after cloud passage may be as effective as evacuation even in severe accident sequences at distances greater than about 10 miles. (Schwartz Testimony, pp. 5-6)

4.1-5 The selection of a radius of about 10 miles for the plume exposure pathway EPZ was made in the NRC's final emergency preparedness regulations published August 19, 1980, which reference NUREG-0396. Therefore the rationale for selection of the plume exposure pathway EPZ described above holds for the Indian Point site. (Id., p. 6)

B. DEVELOPMENT OF THE INDIAN POINT EPZ

4.1-6 The boundary of the EPZ was developed by modifying a circle with a radius of ten miles, in order to preserve or group major population areas; simplify boundary definitions as much as possible (by using, e.g., political divisions or major roads); and recognize important

topographic features, such as rivers, hills and valleys, and utilize them as boundaries when practical. The resulting irregularly shaped EPZ, which generally encompasses an area greater than a ten-mile radius, reflects the consideration and inclusion of the various factors required by NUREG-0396 and NUREG-0654. (Parsons Testimony, p. 15; FEMA Testimony, pp. 36-37) Con Edison Ex. 8 (admitted, Tr. 11507) is a map showing the EPZ boundary.

4.1-7 The drawing of the EPZ boundary was carried out by Parsons Brinckerhoff in consultation with the State, the four counties, the Licensees and other consultants. (Parsons Testimony, p. 15) Philip Schmer, Assistant Director of the Office of Natural Disaster, Civil Defense, Orange County, and Michael Scalpi, Civil Defense Director, Putnam County, confirmed that they and Parsons Brinckerhoff worked closely together to develop all aspects of the Orange and Putnam plans. (Tr. 12074-12079) Donald P. McGuire, Deputy Director, Office of Emergency Services, Rockland County, confirmed that his predecessor had participated in the radiological emergency planning process, including development of the boundary for the EPZ. (Tr. 3788) Lieutenant Governor and former Westchester County Executive Alfred DelBello initially stated that Westchester County had had no input into

emergency planning. (DelBello Testimony, p. 2; Tr. 5868) However, when shown his own written statements to the contrary, he agreed that Westchester County had had extensive input into the development of the Westchester plan. (Tr. 5868-5892)

4.1-8. Donald B. Davidoff, Director of the New York Radiological Emergency Planning Group, and Lawrence B. Czech, Chief, Nuclear Protection Planning, New York Radiological Emergency Planning Group, both expressed unqualified approval of the current Indian Point EPZ. They stated that the generic justification for a 10-mile EPZ has not been successfully attacked in the case of the Indian Point EPZ, and that no specific evidence has been presented justifying its enlargement. (Davidoff/Czech Testimony, p. 15)

4.1-9 Plume modeling characteristics, including both meteorological and plant-specific release characteristics, are conspicuous in their absence from the factors to be considered in selecting an EPZ boundary, both in the formulation of Contention 4.1 and in the underlying regulations. It is obvious that the NRC intended to select an EPZ radius of about 10 miles on a generic basis and to modify it only on the basis of local conditions. (See the San Onofre decision, discussed above.) However, even if

plume modeling were to be considered in determining the EPZ boundary, contrary to NUREG-0396 and NUREG-0654, uncontroverted testimony offered by the licensees shows that even under severe accident conditions, and assuming (unrealistically) that there is no sheltering or evacuation, dosages would be such that, even using WASH-1400-based source terms, more than 95% of the early fatality risk would be confined to a four mile radius from the Indian Point site. (Finding 1.1-274) Intervenor witnesses Beyea and Palenik admitted that NUREG-0396 shows that the frequency of plume deposition falls off sharply beyond 10 miles. (Tr. 3194-3195) Reduction in the source term would further shrink the zone of early fatality risk. (See Findings 1.1-303)

4.1-10 It was clearly established by the L. Cohen Testimony and Mr. Cohen's responses to the Board's questions (Tr. 11714-11738) that even if meteorology were a factor to be considered in drawing an EPZ boundary, there is no basis in the Indian Point meteorology for changing the Indian Point EPZ. Mr. Cohen, a Con Edison Senior Meteorologist with extensive experience and expertise in general and Indian Point site specific meteorology, (See, Con Edison Onsite Testimony, pp. 2-4) explained in detail that the meterorology of Indian Point is influenced by terrain features which produce a valley wind system and

which induce a steering mechanism for low level wind flow (L. Cohen Testimony, p. 2) The combination of meteorology and terrain is such that under differing wind patterns signifi- cant plume transport beyond the current plum exposure EPZ is highly unlikely (Id., pp. 2-3, Tr. 11731-11734). Mr. Cohen's general expertise in meteorology and his detailed familiarity with Indian Point site specific meteorology and terrain stands in contrast to the experience of Mr. Gutman, a witness for New York City Council members. Mr. Gutman is not a professional meteorologist, has never taken a formal course in meteorology and was involved with meteorological issues in only a marginal way during his employment with the Environmental Protection Agency. (Tr. 15165-15167)

C. EMERGENCY RESPONSE OUTSIDE THE EPZ

4.1-11 Existing evacuation planning in the vicinity of Indian Point extends out well beyond the EPZ. Parsons Brinckerhoff examined roads outside of the EPZ, to the next major intersection, to ensure that no bottlenecks outside the EPZ could cause traffic to back up into the EPZ. (Della Rocca, Tr. 11944-11946) Also, the plans call for traffic control measures outside the EPZ to ensure that egress from the EPZ will not be impeded. (Della Rocca, Tr. 11941, 11943-11944) The greater portion of each of the

four counties composing the Indian Point plume exposure pathway EPZ is located outside of the EPZ, and there is no reason why emergency response could not be extended beyond the EPZ boundary if necessary in a specific accident sequence. As noted in NUREG-0654, detailed planning within the 10 mile zone would provide a substantial base for expansion of response efforts in the event that this proved necessary. (NUREG-0654, p. 12) This is particularly so at Indian Point, where expansion of the response area would be done by officials and emergency response organizations already trained and involved in the effort.

4.1-12 A number of intervenor witnesses alleged that their testimony addressed Contention 4.1. See "Index of Witnesses Presenting Testimony on Emergency Planning Issues on Behalf of the Intervenors: UCS, NYPIRG, RCSE, WBCA, Parents, WESPAC, FOE and Audubon and on Behalf of the Attorney General of the State of New York," passim. However, these non-expert witnesses, if indeed they mentioned the size of the EPZ at all, merely recited their unsupported belief that a release from the Indian Point site would necessitate evacuation beyond a ten mile radius. Such assertions are not useful in addressing Contention 4.1.

D. ULTIMATE FINDING

4.1-13. The radius of the EPZ must be about 10 miles. The Indian Point EPZ was developed using criteria contained in NUREG-0396 and NUREG-0654, and in consultation with local governments. It groups major population areas, simplifies boundary definitions, and recognizes important topographic features. Evacuation planning extends beyond the EPZ. The Board therefore finds that the EPZ is adequate as drawn and that Contention 4.1 should not be adopted.

III. CONTENTION 4.2(a)

4.2(a)-1. Contention 4.2(a) reads:

> The following specific, feasible off-site measuresshould be taken to protect the public: (a) Potassium iodide should be provided in an appropriate form for all residents in the EPZ.

4.2(a)-2 Iodine 131 may be one of the most harmful components of a release, because, when ingested, it concentrates in the thyroid, increasing the likelihood of thyroid cancer and thyroid nodules. (NYS Ex. 2, admitted Tr. 11313, pp. 84-87, prepared with the direct involvement of New York State witness Davidoff, Tr. 11293-11294)

4.2(a)-3 Potassium iodide protects the thyroid from the potentially harmful effects of iodine 131 through two mechanisms: loading the gland with stable iodine so that it cannot absorb any ingested iodine 131, and serving as an auto-regulatory mechanism in the thyroid, limiting the rate at which further iodine can be accumulated by the gland. (Id., p. 87)

4.2(a)-4 Potassium iodide is available at the Indian Point site, for use of persons at the site. (Con Edison Onsite Emergency Plan, Section 6.4.2; Section 10, Appendix E; Power Authority Onsite Emergency Plan, Sections 6.5.1, 6.5.2; Section 10, Appendix F)

4.2(a)-5 During the March 9, 1983 exercise, adequate provisions were made for distribution of potassium iodide if necessary. (FEMA Post Exercise Assessment, April 14, 1983, pp. 32, 43, 50-51, 58)

4.2(a)-6 Distribution of potassium iodide to the general population in the EPZ is opposed by public health experts, for a number of reasons. Blocking of the uptake of iodine 131 by the thyroid means that the isotope remains in the bloodstream, thus exposing the blood-forming organs to an increased dose of radiation. As a result, the risk of premature death due to bone marrow suppression or leukemia could be significantly increased. (NYS Ex. 2, p. 87) Side effects are possible. (Id., p. 88) The drug has a relatively short shelf life of only three years, (id., p. 89), and deteriorates quickly when exposed to light (Id.), or moisture. (Curran, Tr. 5188) Possession of potassium iodide by the general population could well cause a "panacea effect," in which persons who have taken the drug believe they are protected from all radiation effects and thus expose themselves unnecessarily to radiation. (Solon, Tr. 9085-9086) Moreover, potassium iodide can be manufactured very quickly and easily. (Solon, Tr. 9075) For these and other reasons, the Committee on Public Health of the New York Academy of Medicine has recommended against the stockpiling of potassium iodide for possible use in New

York City. (PA Ex. 30, admitted, Tr. 9079) The New York State Disaster Preparedness Commission has come to the same conclusion for all segments of the general population. (Davidoff/Czech Testimony, p. 16; NYS Ex. 2, p. 89)

4.2(a)-7 It may be advisable to stockpile potassium iodide for certain classes of offsite persons, such as emergency workers and persons who cannot be evacuated (e.g., prisoners, patients). The New York Radiological Preparedness Group has been advised that this is being done. (Davidoff/Czech Supplemental Testimony, p. 12)

4.2(a)-8 Interestingly, the only intervenor witnesses who presented more than merely conclusory testimony on potassium iodide called for distribution of the drug outside the EPZ, in New York City. Dr. Leonard Solon, Director, Bureau of Radiation Control, New York City Department of Health, gave it as his own personal, unofficial view that potassium iodide should be stockpiled for use in New York City. (Solon Testimony, pp. 1, 5-6) He did not call for distribution of the drug to New York City residents. (Tr. 9088) However, he admitted that there was an "unfortunate" lack of human data on the effects of the drug (Tr. 9081-9082), that the cost of stockpiling it in New York City would be at least $13

million, reincurred every two years because of the short shelf life of the drug, and that there could be a panacea problem. (Tr. 9085-9086) He disagreed with the conclusion of the New York State Health Department and the New York Academy of Medicine that the drug should not be distributed to the public, even though he had not examined the reasons for the Health Department's recommendation and acknowledged the expertise of the Academy of Medicine. (Solon Testimony, pp. 5-6; Tr. 9061-9062, 9077) Finally, his testimony is based on meteorological assumptions as to the accuracy of which he is admittedly incompetent. (Tr. 9054-9055)

4.2(a)-9 Many other intervenor witness asserted that their testimony addressed Contention 4.2. See "Index of Witnesses Presenting Testimony on Emergency Planning Issues on Behalf of the Intervenors: UCS, NYPIRG, RCSE, WBCA, Parents, WESPAC, FOE, and Audubon and on behalf of the Attorney General of the State of New York," passim. However, these non-expert witnesses, insofar as they mentioned potassium iodide at all, merely stated their unsupported belief that it should be distributed to all persons in the EPZ. Such conclusory assertions are not useful in addressing Contention 4.2(a).

A. ULTIMATE FINDING

4.2(a)-10 Distribution of potassium iodide to the general populace within the EPZ is opposed by public health experts and the State of New York, because it may have harmful as well as beneficial effects, has a short shelf life, requires careful storage, and could produce a panacea effect. The Board agrees with this analysis and therefore finds that the drug should not be distributed to all residents in the EPZ, and that accordingly Contention 4.2(a) should not be adopted.

IV. CONTENTION 4.2(b)

4.2(b)-1 Contention 4.2(b) reads as follows:

> The following specific, feasible off-site measures should be taken to protect the public: . . . (b) Adequate sheltering capability should be provided for all residents in the EPZ.

4.2(b)-2 Sheltering, in the context of a radiological emergency caused by a release from a nuclear power plant, does not require civil defense/nuclear attack shelters with their blast and fallout protection. Instead, it merely requires going indoors and controlling ventilation, as by closing doors and windows and turning off air conditioners. (Davidoff/Czech Testimony, p. 16; Tr. 11356-11357) The joint NRC/EPA task force recommended in NUREG-0396 that no specific fallout shelters be constructed. (FEMA Testimony, p. 38) NUREG-0654 takes the same position. (Id., pp. 39-40) There are sufficient structures presently in the EPZ to shelter the general population. (Id., p. 39)

4.2(b)-3 The emergency plans contain a sheltering option and guidelines for decision-makers. Orange County RERP, § III.H; Westchester County RERP, § III.H; Putnam County RERP, § III.H; NYS Ex. 10, Draft RERP produced by Rockland County, § III.H) The emergency planning brochure devotes an entire page to sheltering

instructions. (UCS/NYPIRG Ex. 14, admitted, Tr. 10335, p. 5) Further instructions could be issued over the EBS during an emergency. (Id.)

4.2(b)-4 County plans make use of public fallout shelters to supplement sheltering in houses, places of work, schools, and similar facilities. (Davidoff/Czech Testimony, p. 16; FEMA Testimony, p. 39)

4.2(b)-5 Houses in this area of the country are designed such that they can be made nearly airtight, in order to save on winter heating bills. They therefore make particularly effective shelters. (Czech, Tr. 11356-11357; Keller, Tr. 15076-15077)

4.2(b)-6 The principal element of sheltering is minimizing air exchange, and because of the efficiency of buildings in the Indian Point area in doing so, it was not necessary to survey the sheltering capability of buildings in order to conclude that sheltering was a viable protective action within the EPZ. (Davidoff/Czech Testimony, pp. 16-17; Tr. 11355-11358) However, a sheltering survey has been conducted by the Army engineers, which found facilities adequate for war conditions, where there would be higher potential radiation levels than in a release from Indian Point. (Schmer, Tr. 12140-12141)

4.2(b)-7 The resources necessary to implement sheltering are an alert system (sirens) and a notification system (the Emergency Broadcast System). The EBS worked well during the March 3, 1982 exercise (FEMA Post Exercise Assessment, May 27, 1982, pp. 30-31, 42, 51, 59-60; Sears, Tr. 12240-12242; D. McGuire, Tr. 4074) and the March 9, 1983 exercise. (FEMA Post Exercise Assessment, April 14, 1983, pp. 26-27, 39-40, 47-48, 53-54) All sirens operated properly during the March 9, 1983 exercise. (Id., Jackson, Tr. 11759; Sears, Tr. 12240-12242) Even Rockland County would be able to implement sheltering, despite the county's withdrawal from the emergency planning process. (D. McGuire, Tr. 4073-4074)

4.2(b)-8 UCS/NYPIRG witnesses Beyea and Palenik admitted that sheltering was a viable alternative within the EPZ for those for whom evacuation is not attempted or is not successful. (Beyea/Palenik Testimony, pp. 47-48)

4.2(b)-9 Many intervenor witnesses alleged that their testimony addressed Contention 4.2. See "Index of Witnesses Presenting Testimony on Emergency Planning Issues on Behalf of the Intervenors: UCS, NYPIRG, RCSE, WBCA, Parents, WESPAC, FOE, and Audubon and on Behalf of the Attorney General of the State of New York," passim. However, these non-expert witnesses, if indeed they

mentioned sheltering at all, merely stated that houses in the Indian Point area were, in their opinion, unable to provide adequate sheltering. Such conclusory assertions are inadequate to overcome expert testimony and are not useful in addressing Contention 4.2(b).

ULTIMATE FINDING

4.2(b)-10 Sheltering, in the context of a nuclear power plant accident, involves going indoors and controlling ventilation. On the strength of the evidence presented by FEMA, the State of New York and the NRC Staff, and in the absence of evidence to the contrary, the Board finds that adequate sheltering capacity presently exists within the EPZ to protect the general population. The Board further finds that the present sirens and EBS are adequate to implement sheltering. Assuming arguendo that sheltering capacity did not exist, the Board finds that the essential elements of specificity and feasibility under Commission Question 4 have not been satisfied, because the intervenors presented no evidence on these subjects. The Board therefore finds that Contention 4.2(b) should not be adopted.

V. CONTENTION 4.2(c)

4.2(c)-1 Contention 4.2(c) reads as follows:

> The following specific, feasible off-site measures should be taken to protect the public: . . . (c) License conditions should prohibit power operation of Units 2 and 3 when the roadway network becomes degraded because of adverse weather conditions.

4.2(c)-2 Contention 4.2(c) does not suggest an off-site measure that should be taken to protect the public. Rather, it suggests an amendment to the operating licenses of the Indian Point units. As such, it is not a proper contention under Commission Question 4.

4.2(c)-3 The emergency plans for Indian Point do not call for evacuation alone, but rather set forth a range of protective actions, of which general evacuation is only one option. In any case in which road conditions are bad, the initial recommendation would be to take shelter. In case evacuation may not be feasible due to adverse weather, more specific recommendations (e.g., shelter in basement, ad hoc respiratory protection) can be broadcast to the public. (Sears Testimony, p. 54)

4.2(c)-4 The evacuation time estimates contained in the county plans are given in ranges that take into account the possibility of degraded roads due to adverse weather conditions. Local officials can therefore take

this factor into account when making subsequent recommendations after the initial recommendation to take shelter. (Id., p. 53)

4.2(c)-5 While it is true that a very heavy snowstorm can reduce road capacity, the licensees' consultants took this into account in developing their evacuation time estimates. (Della Rocca, Tr. 11946-11947)

4.2(c)-6 Authorities in the area have many snowplows, and are used to dealing with snow. (Sears, Tr. 12343-12344) After the heavy snowstorm known as the "Blizzard of 1983," abandoned cars were cleared from the highways by 8:00 or 9:00 the next morning. (Schmer/Scalpi, Tr. 12198)

4.2(c)-7 A successful evacuation is possible in all weather conditions. (Davidoff/Czech Testimony, p. 17) At least two of the counties have equipment to keep roads open in a snowstorm, and in addition have commitments from the State for additional equipment if needed. (Schmer,Scalpi, Tr. 12116)

4.2(c)-8 If roads are blocked due to a heavy snowstorm, that same storm will in all probability pull the release of radioactive material down close to the plant. (Sears, Tr. 12342; Blond, Tr. 12953-12957) Current

research indicates that snow may have a substantially greater effect than rain in pulling down a plume. (Blond, Tr. 12957) A release in cold weather would form a cloud, causing self-rain-out that would also pull down the plume very close to the plant. (Sears, Tr. 12343-12344; Blond, Tr. 12954-12956) The self-rain-out phenomenon was actually observed at the Ginna plant last year. (Blond, Tr. 12954)

4.2(c)-9 Any reduction in risk to the public by forcing the Indian Point units to shut down during adverse weather will be offset by an increase in risk caused by two factors: (a) frequent shutdown and startup transients would cause increased wear and tear on plant components, and (b) electric grid reliability would be reduced if the plants were shut down. (Sears Testimony, pp. 54-55)

4.2(c)-10 For severe, fast-moving accident scenarios, the preferred protective action, regardless of road conditions, would always be sheltering until after plume passage and then later relocation. Thus reduction in individual risk would not depend on calculated evacuation times. (Id.) A slow-moving scenario would probably not result in a release in less than 12 hours, and more likely 24 hours. (See, Findings 3.9-6 through 3.9-8) Thus there would probably be time to clear the roads. If there is not, the alternative is the same as with a fast-moving accident -- sheltering followed by relocation.

A. ULTIMATE FINDING

4.2(c)-11 No evidence was offered that meteorological conditions in the Indian Point area warrant imposition of the requirement suggested by Contention 4.2(c). Rather, the credible evidence offered suggests that the capability to respond to adverse weather exists in the area. Sheltering followed by relocation is an alternative emergency response available to government officials if road conditions are degraded. The evacuation time estimates take road conditions into account. Principal roads can be quickly cleared. Storm conditions will pull any plume down close to the plant. Finally, Contention 4.2(c) is not a proper contention under Commission Question 4, because it does not suggest an offsite measure. The Board therefore finds that Contention 4.2(c) should not be adopted.

VI. CONTENTION 4.2(d)

4.2(d)-1 Contention 4.2(d) reads as follows:

> The following specific, feasible off-site procedures should be taken to protect the public . . . (d) The roadway network should be upgraded to permit successful evacuation of all residents in the EPZ before plume arrival time.

4.2(d)-2 Issues relating to Contention 4.2(d) are very similar to those discussed under Contention 3.9 and portions of Contention 3.3. The basic issues are discussed under those contentions.

4.2(d)-3 The road network within the EPZ is presently capable of handling large volumes of traffic. For instance, during ceremonies at West Point welcoming the Iranian hostages home, the roads accommodated without difficulty a volume of traffic exceeding expected evacuation loads. (Schmer, Tr. 11212-11213)

4.2(d)-4 As discussed at length under Contention 3.3, the evacuation time estimates prepared by Parsons Brinckerhoff Quade & Douglas are based on assessments of existing road capacities and conservative assumptions about usage levels.

4.2(d)-5 Officials of Westchester, Orange and Putnam Counties admitted that roads are presently adequate

for timely evacuation (Del Bello, Tr. 5984-5985; Jurkowski, Tr. 5293-5294; Schmer/Scalpi, Tr. 12123-12124, 12142-12143), and a Rockland County official was unwilling to take a position. (Gdanski, Tr. 3547-3549, 3553-3554)

4.2(d)-6 Roadway improvements are planned or underway in the EPZ. Route 9 is being widened near Peekskill, Route 301 will be improved, and a number of other roads in Putnam County are scheduled for improvement. (Scalpi, Tr. 12143)

4.2(d)-7 The present road system in the four counties does not need upgrading because timely evacuation is possible with the existing network. (FEMA Testimony, p. 17) FEMA based this testimony on four separate evacuation time studies: one by the New York State Office of Disaster Preparedness and the New York State Department of Health; the Parsons Brinckerhoff study; a study done for FEMA by CONSAD Research Corporation; and a study done for FEMA by Dr. Thomas Urbanik. (McIntire, Tr. 1448)

4.2(d)-8 As discussed in Findings 3.9-6 through 3.9-8 the alleged problem with road adequacy does not actually exist. In the fast-developing "V" sequence accident there will not be time to evacuate before plume arrival, regardless of road improvements, and there is no

significant increase in consequences due to this. In the slow-developing "2RW" sequence accident there is ample time to evacuate on existing roads.

4.2(d) 9 Many intervenor witnesses asserted that their testimony addressed Contention 4.2. See "Index of Witnesses Presenting Testimony on Emergency Planning Issues on Behalf of the Intervenors: UCS, NYPIRG, RCSE, WBCA, Parents, WESPAC, FOE, and Audubon and on behalf of the Attorney General of the State of New York," passim. However, these non-expert witnesses, insofar as they mentioned upgrading of the road system at all, merely stated their unsupported belief that roads should be upgraded. None of these witnesses addressed the underlying issues of Commission Question 4 -- the specific upgrading recommended, or its feasibility. Such conclusory assertions, unaccompanied by any evaluation of the impact of road upgrading on evacuation or accident consequences, are not useful in addressing Contention 4.2(d).

A. ULTIMATE FINDING

4.2(d)-10 Evacuation time estimates prepared by the licensees' consultants are accurate. FEMA, New York State and local government officials agree that effective evacuation is possible with the existing road network.

This is particularly so in view of the likely accident sequences. The Board therefore finds that no case has been made for the adoption of Contention 4.2(d). Finally, in its recent decision in Southern California Edison Co. (San Onofre Nuclear Generating Station, Units 2 and 3), CLI-83-10 (April 4, 1983), the NRC stated that its emergency planning regulations contemplated the use of existing offsite resources and did not comprehend the building of new hospital capacity to treat persons who might be exposed to high doses of radioactivity that might result from a release from the San Onofre plant. On the basis of this decision, the Board finds that major offsite construction projects, such as the general upgrading of the road network contemplated under Contention 4.2(d), are not proper issues to be considered in this proceeding.

VII CONTENTION 4.7

4.7-1 Contention 4.7 reads as follows:

> The emergency plans should be upgraded to provide more adequate methods for alerting and informing persons who are deaf, blind, too young to understand the instructions, or who do not speak English.

4.7-2 Only one witness suggested a solution (the TTY system) to the problem of alerting deaf persons, but he admitted that very few people have this sophistica- ted equipment. (Rowley Testimony, p. 2)

4.7-3 No intervenor witnesses asserted that the blind cannot be alerted by sirens or listen to radios.

4.7-4 Children too young to understand instructions are universally supervised by adults. Such children would not be passed over in an evacuation because of their inability to understand instructions. Children therefore will receive instructions from parents or other adults who have charge of them.

4.7-5 There are within the EPZ approximately 6000 persons who have identified themselves as being of Hispanic origin, comprising approximately 1900 families. (O'Lear Testimony, pp. 1-2) However, there is no indication of how many of these persons do not speak

English (O'Lear, Tr. 1839; Holland, Tr. 1819-1820), or do not reside with a bilingual person. (O'Lear, Tr. 1824) The Spanish language newspapers El Tiempo and El Diario (which the Board takes notice are the principal Spanish language newspapers in the area) have no distribution within the EPZ. (UCS/NYPIRG Ex. 16, admitted, Tr. 12481)

4.7-6 FEMA originally expressed concern about the lack of emergency planning booklets and EBS announcements in Spanish. (FEMA Testimony, pp. 15, 45; FEMA Post Exercise Assessment, May 27, 1982, p. 42) After further study, however, both FEMA and the State of New York concluded that there was no need to distribute materials in the EPZ in languages other than English, and that it would be impractical to do so. (FEMA Supplemental Testimony, p. 5; Davidoff/Czech Testimony, p. 21; Davidoff/Czech Supplemental Testimony, p. 13)

4.7-7 The Board takes notice that persons who do not speak English can nevertheless be alerted by sirens, and can receive interpretations of EBS messages from English-speaking friends.

4.7-8 Finally, the police department in Haverstraw, the most heavily Hispanic community in the EPZ, has Spanish-speaking officers who would assist in a radiological emergency. (Holland, Tr. 1691-1692)

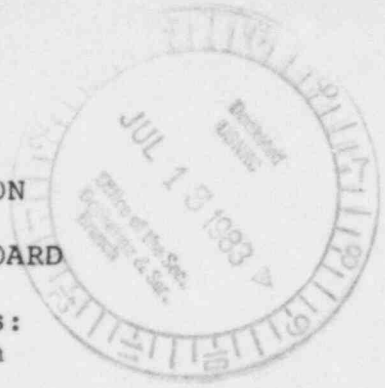
4.7-9 Persons with special needs, such as the deaf, the blind, and those too young to understand instructions, are provided for in emergency situations by friends and relatives, who have major responsibility for their care. (Davidoff/Czech Testimony, pp. 18, 21) Interpersonal relationships (family and friends) are one of the most important factors in how and what individuals learn of an impending threat, and what actions they take. (Smith Testimony, pp. 8-9) The State of New York is developing a series of informative documents for such persons, using the expertise of the State Department of Health, which has extensive experience in conducting such programs. (Davidoff/Czech Testimony, p. 19) Significant progress is being made in planning for the needs of those who do not speak English by enlisting the aid of community social and religious leaders. (FEMA Supplemental Testimony, pp. 3, 5)

4.7-10 A substantial number of intervenor witnesses alleged that their testimony addressed Contention 4.7. See "Index of Witnesses Presenting Testimony on Emergency Planning Issues on Behalf of the Intervenors: UCS, NYPIRG, RCSE, WBCA, Parents, WESPAC, FOE, and Audubon and on Behalf of the Attorney General of the State of New York," passim. Perhaps because Contention 4.7 was reformulated and narrowed in scope after this testimony was

submitted, the testimony bears no relation to Contention 4.7.

A. ULTIMATE FINDING

4.7-11 The Board finds that adequate means already exist for alerting and informing persons who are deaf, blind, too young to understand instructions, or do not speak English. Moreover, no specific, feasible measures have been proposed to upgrade existing plans. Therefore, Contention 4.7 should not be adopted.



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) : July 11, 1983

CON EDISON'S PROPOSED FINDINGS OF FACT REGARDING COMMISSION QUESTION 5

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COMMISSION QUESTION 5

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COMMISSION QUESTION 5

I. INTRODUCTION

5.0-1 Commission Question 5 reads as follows:

> 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).

5.0-2 Risk, whether expressed on an individual or a societal basis, can be measured by several health and economic indices. These indices are affected differently by population distribution and specific plant features. In comparing risks among plants, Licensees and Staff have emphasized early and latent fatality risks, as did the Commission in formulating its preliminary safety goals. (48 Fed. Reg. 10722 (1983); Paddleford, et al.; Rowsome/Blond 5)

5.0-3 Commission Question 5 is addressed in these findings from the following perspectives: (1) a comparison of the risks from Indian Point Units 2 and 3 to the Commission's preliminary safety goals; (2) the risks from Indian Point Units 2 and 3, based upon the IPPSS,

compared to the results of site and plant specific PRAs of a number of other nuclear power plants; (3) a comparison of Indian Point risk with results of the NRC Task Force Report (NUREG-0715); and (4) a discussion of the relevance of the Sandia Siting Study.

II. COMPARISONS WITH NRC PRELIMINARY SAFETY GOALS

5.0-4 The first approach taken to address Commission Question 5 is to compare the risks of early and latent fatalities and core melt frequency at the Indian Point power plants to the Commission's preliminary safety goals published in the March 14, 1983 "Policy Statement on Safety Goals for the Operation of Nuclear Power Plants", 48 Fed. Reg. 10772 (1983). (Paddleford, et al., pp. 2-3, 4-8; Rowsome/Blond 5, pp. B-8 to B-20) The Commission's preliminary safety goals are stated as quantitative design objectives which apply to both individual and societal risk. (Id.)

5.0-5 The safety goal policy statement makes it clear that the goals are not to be used to make licensing decisions nor are they intended to be a threshold of minimally acceptable risk. (Rowsome/Blond 5, pp. B-9 to B-10) However, the preliminary safety goals represent a national benchmark in terms of design objectives against which all

nuclear power plants can be compared. Therefore, the comparison of the Licensees' and Staff risk results for the Indian Point plants to the preliminary safety goals is one method of determining if these plants are within the range of risks posed by other nuclear power plants. (Paddleford, et al., p. 4) Such comparisons are illuminating for purposes of the case at hand. (Rowsome/Blond 5, p. B-9)

5.0-6 In terms of individual risk, the Commission's preliminary safety goals state that the early fatality risk to an average individual in the vicinity (a one-mile radius) of a nuclear power plant should not exceed 1/10th of one percent of the sum of early fatality risk to that individual from other accidents. (Paddleford, et al., p. 5; Rowsome/Blond 5, p. B-11) The Licensees calculated this risk for Indian Point to be well within the preliminary safety goal by a factor of approximately 70 for Indian Point 2 and a factor of approximately 75 for Indian Point 3. (Paddleford, et al., p. 6) Even if the admittedly conservative Staff consequence analysis, (see Findings 1.1-243, 1.1-258, 1.1-308, 1.1-364), is used as a basis for comparing Indian Point risk to the safety goals, the Staff's assessment of the plants in their present configuration, for a variety of emergency response and medical treatment assumptions, also concluded that the

estimates of individual risk consistently meet the safety goal design objective for early fatalities. (Rowsome/Blond 5, pp. B-12, 13) Further, the Staff calculations combine the risk from Indian Point 2 and 3. Thus, the risk from either unit would be only a portion of the computed total. (See Rowsome/Blond 5, p. B-12, n.3)

5.0-7 For societal risk, the Commission's preliminary safety goal is that the latent cancer fatality risk to the population in the vicinity (within 50 miles) of a nuclear power plant should be less than 1/10th of one percent of the cancer fatality risks from other causes. (Paddleford, et al., p. 6; Rowsome/Blond 5, p. B-14) Comparing the average background latent cancer fatality risk within a 50-mile radius of the Indian Point operation of the facilities, the Licensees calculated that the societal risk from Indian Point is well within the safety goal by a factor of approximately 165 for Indian Point Unit 2 and 710 for Indian Point Unit 3. (Paddleford, et al., p. 6) Once again, based upon its more conservative analysis, the Staff agreed that the projected risk of cancer fatalities per year for the Indian Point site meet, by more than a factor of 10, the design objective in the safety goal policy statement. (Rowsome/Blond 5, p. B-16 to B-17)

5.0-8 Uncertainty in the respective risk analyses of the Staff and Licensees does not appear to greatly affect the determination regarding whether the Indian Point facilities meet the Commission's quantitative safety goals for individual and societal risk. When the Licensees risk assessment, which the Staff agrees is in certain respects more realistic than their own, (see Findings 1.1-30. 1.1-198), is used to determine the degree of conformance with the preliminary safety goals, the uncertainties in the calculated health risks for Indian Point are offset by the large margins between these calculated risks and the preliminary safety goals. (Paddleford, et al., pp. 4-5) Further, Licensees' safety goal calculations are point estimates based upon mean release frequencies, WASH-1400 equivalent source terms, and S1 consequence analysis (dose multiplier of 1). (Potter, Bley, Tr. 12672-12673; Potter, Tr. 12691-12693) Therefore, most uncertainty would be on the low side. (See Licensees 1, p. 109)

5.0-9 While the uncertainties surrounding the Staff estimates of early fatality risk are larger than the margin by which that risk meets the design objective, the Staff testified that treatment of uncertainties in its consequence analysis under Commission Question 1 suggests

that the risk of early fatalities originating from severe reactor accidents at Indian Point is quite unlikely to represent more than a few tenths of one percent of the background non-nuclear risks for those in the site vicinity, and it is quite likely to be well below 1/10th of one percent. (Rowsome/Blond 5, p. B-13 to B-14)

5.0-10 Uncertainties with regard to the latent cancer fatality risk calculated by the Staff in its conservative analysis would not affect at all the conclusion that this risk meets the quantitative design objectives of the safety goals. This is because the margin between the Staff's calculated risk and the safety goal clearly demonstrates that it is extremely unlikely that the Indian Point units, operating together, pose anywhere near 1/10th of one percent of the background of cancer fatalities to the average individual within 50 miles of the site. (Rowsome/Blond 5, p. B-16)

5.0-11 Both the Staff and the Licensees' testimony indicated that it is likely that a significant reduction in the source terms for severe reactor accidents may be forthcoming in the not too distant future. (See Findings 1.1-276 to 1.1-312) Such reduced source terms will result in even larger margins between the assessed early

fatality and latent cancer fatality risk for Indian Point and the Commission's preliminary safety goals for those risk indices. (Paddleford, et al., p. 5)

5.0-12 Although core melt frequency is a poor indicator of public health risk at Indian Point, (see e.g. Finding 1.1-127), the Staff and Licensees also compared their calculated median core melt frequencies with the Commission's secondary quantitative design objective of one in ten thousand ($1x10^{-4}$) per year of reactor operation. The Licensees' assessment, reflecting current plant configurations, shows a core melt frequency from both internal and external initiators for Indian Point 3 that meets the Commission's preliminary safety goal, and for Unit 2 a frequency which, when rounded off, is exactly equal to the safety goal. (Paddleford, et al., p. 7; Bley, Tr. 12681) When comparisons are made between the safety goal and median core melt frequencies from internal initiators only, both plants clearly meet the goal. (Paddleford, et al., p. 7) When Indian Point is compared to other plants, this internal only comparison is more valid because, other than the IPPSS and the Zion PRA, no PRAs have detailed and comprehensive analyses of external events. (See Paddleford, et al., pp. 6, 8)

5.0-13 Although the Staff calculated overall core melt frequencies for both Indian Point facilities at 3.5×10^{-4} per year, which is slightly above the design objective, there were acknowledged conservatisms and substantial uncertainties in the Staff's analysis. For example, no credit was given for strengthening of the ceiling panels in the control room of either unit, nor did the Staff eliminate the conservatisms in its calculation of the frequency of hurricane induced core melt events. (Rowsome/Blond 5, pp. B-18 to B-19) Further, Staff witness Rowsome testified that, in any case, more than half of the operating nuclear plants in the United States may not meet the preliminary safety goal for core melt frequency. (Tr. 12878-12879)

5.0-14 The frequency of severe release categories is a far better measure of offsite radiological risk than the overall core melt frequency. The frequency of core melts leading to the more severe release categories which account for nearly all of the offsite radiological risk is well below the quantitative safety goal of 10^{-4} per reactor year. (Rowsome/Blond 5, pp. B-19 to B-20; Rowsome, Tr. 12912)

III. COMPARISONS OF IPPSS RESULTS WITH OTHER PRAs

5.0-15 Another way to compare the risks posed by the Indian Point plants with those posed by other nuclear 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. (Paddleford, et al., p. 9)

5.0-16 The parties agree that the thirteen or fourteen PRAs which have been published to date contain differences in scope, consistency of approach, adequate data basis, methodology and general sophistication. (Paddleford, et al., pp. 9-10; Rowsome/Blond 5, p. A-4; Sholly 5, pp. 14-22) This is because the state of the art in PRA methodology has been evolving rapidly and continuously over the last ten years. (Paddleford, et al., p. 9)

5.0-17 Only the IPPSS and Zion PRAs include a detailed and comprehensive treatment of the contribution of external events such as earthquakes, winds, fires and floods to the risk from operation of nuclear plants. Therefore, appropriate comparisons with other risk assessments are on the basis of internally initiated events

only. (Paddleford, et al., p. 9; Rowsome/Blond 5, pp. A-6 to A-7) While the difficulties identified above limit the ability to make overall comparisons between the other published PRAs and the IPPSS, all of the published studies have analyzed in roughly the same manner the so-called "internally" initiated events. (Rowsome/Blond 5, p. A-6) Therefore, the Licensees and Staff utilized the PRAs done for the following plants to compare risk from internally initiated events at those plants with those at Indian Point: ANO-1, Big Rock, Calvert Cliffs, Crystal River, Grand Gulf, Limerick, Oconee, Peach Bottom, Sequoyah, Surry and Zion. (Rowsome/Blond 5, p. A-5; Paddleford, et al., p. 11) The Licensees made additional comparisons between Indian Point and Staff-generated PRAs for the Millstone and Browns Ferry facilities, and the German Biblis reactor. Data from a number of plant specific PRAs, as compiled by the Staff, were used to make comparisons of 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 the plant. (Paddleford, et al., pp. 10-11)

5.0-18 The PRAs selected for comparison generally reflect the full range of designs of nuclear power plants that have been built, high and low population densi-

ty sites, a variety of power levels, and various principal reactor vendors and architect/engineers. (Paddleford, et al., p. 10; Rowsome/Blond 5, p. A-6)

5.0-19 The internally initiated events for which comparisons were made by the Licensees and NRC staff encompass spontaneous loss of coolant accidents, spontaneous transient events, and loss of coolant accidents induced by spontaneous transients. Loss of offsite power has also traditionally been considered an internally initiated event. (Rowsome/Blond 5, p. A-6)

5.0-20 Some of the PRAs used in the comparisons by the Licensees and Staff, i.e., Grand Gulf, Sequoyah, Calvert Cliffs and Oconee, were performed as part of the NRC Staff's "Reactor Safety Study Methodology Applications Program" (RSSMAP). Others, i.e., Browns Ferry, Arkansas Nuclear One (ANO), Crystal River and Millstone, were prepared in conjunction with the NRC sponsored "Interim Reliability Evaluation Program" (IREP). The Surry and Peach Bottom units were evaluated in the Reactor Safety Study and the rest of the plants for which comparisons were made were analyzed in industry-sponsored PRA studies. (Paddleford, et al., p. 11)

A. FREQUENCY OF CORE MELT AND MAJOR RELEASE

5.0-21 Based on a comparison of the PRAs published to date, the "internal" core melt frequencies for the Indian Point units are about average with respect to other plants. (Rowsome/Blond 5, p. A-7 and Fig. 1; Paddleford, et al., p. 12)

5.0-22 With regard to the uncertainties present in the frequencies of internally initiated core melts among the various plants for which the comparisons are made, the smallest uncertainty band is about one order of magnitude, and the largest uncertainty band is over three orders of magnitude. The Indian Point uncertainty for the frequency of internally initiated core melt events is within the range of uncertainty bounds of the other power plants. (Rowsome/Blond 5, p. A-8 and Fig. 2)

5.0-23 Overall frequency of core melt for the Indian Point units is in the range of 10^{-4} per reactor year. (Paddleford, et al., Table 1, p. 7; Rowsome/Blond 5, p. A-14) Results from risk assessments of other plants range from 10^{-3} to about 10^{-5} per year, with many clustering around the values for Indian Point. This is not, however, a fair comparison for two reasons: (1) the Indi-

an Point risk assessments are more comprehensive than the others; had the Indian Point studies been limited to the same scope (i.e., internal events only) as the others, lower core melt frequencies would have been predicted; and (2) not all core melts contribute significantly to offsite radiological risks; many core melt sequences are well contained. (Rowsome/Blond 5, p. A-14)

5.0-24 The Licensees and Staff compared the estimated frequency of a severe release of radioactive material to the environment from internally initiated events for the published PRAs. A severe release is defined as a release which has the potential to cause offsite doses which approach lethal levels. (Rowsome/Blond 5, p. A-8) The frequency of severe release is a much more relevant measure of risk to the public health and safety than the frequency of core melt when one is trying to compare differences between plants. The number of people in the vicinity of the site is only of critical importance should such a severe release occur. (Id., p. A-9)

5.0-25 Comparisons of "internal" severe release frequencies demonstrate that the Indian Point units are among the best (have the lowest severe release frequencies) in comparison to the other power plants. (Rowsome/Blond 5, p. A-9; Paddleford, et al., pp. 23, 25)

5.0-26 The extremely low "internal" severe release frequencies for the Indian Point plants in comparison to other nuclear facilities is a reflection of the capabilities of the engineered safety and other design features, which have been incorporated into the Indian Point plant designs to supplement standard safety features. These features were highlighted in the Director's Order of February 11, 1980. (Rowsome/Blond 5, p. A-9; Paddleford, et al., p. 20)

5.0-27 Among the features that are important in establishing a lower frequency 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 large volume of 2.6×10^6 cu. ft., give 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 the Indian Point site 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 or 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 overpressurization failure. Two recirculation pumps, located inside containment, are unique to Indian Point and are two of the pumps capable of providing containment spray recirculation.

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. In addition to the three emergency diesel generators, 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 insure 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 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, any of three component cooling 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 risks from internal initiating events at Indian Point.

(Paddleford, et al., pp. 20-22; Bley, Tr. 12686)

5.0-28 The frequency of late overpressurization 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 overpressurization 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 overpressurization containment failures. While not specifically quantified, the other design features discussed above would certainly provide further risk reduction. (Paddleford, et al., pp. 22-23)

5.0-29 The low frequency of occurrence of internally initiated severe accident releases is confirmed by an examination of the accident sequences that have been found to lead potentially to high risk for pressurized water reactors such as Indian Point. These are (1) uncontained interfacing systems loss of coolant accidents,

(2) loss of all offsite and onsite AC power and failure of the auxiliary feed water system, and (3) special plant-specific vulnerabilities. (Rowsome/Blond 5, p. A-10)

5.0-30 The Indian Point units are less susceptible to the interfacing systems LOCA for two reasons. First, most pressurized water reactors have three to six large pipes that lead from the reactor to low pressure systems outside containment. The Indian Point units have only one. Second, the Indian Point units have pressure boundary valves on the one line potentially subject to uncontained loss of coolant accidents, the design of which is as reliable as any in nuclear power plants. (Rowsome/Blond 5, p. A-12)

5.0-31 The loss of all AC power and failure of auxiliary feed water are important to risk because virtually all methods of cooling the reactor fuel or removing the heat from the containment require electric power to run motor-driven pumps. The one exception is the steam turbine driven auxiliary feedwater pump. The principal advantage of the Indian Point units over other nuclear facilities is the presence of three gas turbine driven electrical generators as well as three emergency diesel generators per unit to supply essential AC power require-

ments. No other reactor plan has so many backup sources of AC power. Many have only two diesel generators. (Rowsome/Blond 5, p. A-12)

5.0-32 Special plant-specific vulnerabilities have previously been found in reactor designs that met the NRC's licensing requirements, but which were nonetheless vulnerable to severe accidents of one kind or another. Risk assessments are capable of revealing design deficiencies that have been missed by conventional licensing safety analysis and are important to risk. The Indian Point units have been subject to one of the most thorough risk assessments ever done on a nuclear plant and no such plant specific design deficiencies of importance to risk have been found for "internally" initiated accidents. (Rowsome/Blond 5, pp. A-13 to A-14)

5.0-33 The Staff also compared the Indian Point units to other nuclear facilities with regard to their ability to prevent containment failure prior to a core melt from a number of scenarios including overpressurization failure, uncontained interfacing systems LOCAs, earthquake-induced collapse of containment, internal and external missiles, open containment penetrations and steam generator tube ruptures with a stuck open atmospheric

steam release valve. With regard to earthquake-induced failures of the containment, the testimony under Commission Question 1 has indicated that, based upon a reanalysis of the fragility of the Indian Point containments, such an earthquake-induced failure is not possible given the Indian Point containment design at this particular site. (See Finding 1.1-183) With regard to each of the other possible failure modes of the containment considered by the Staff, the Indian Point plants were found to be average or better than average compared to other nuclear plants in their ability to prevent these scenarios. (Rowsome/Blond 5, pp. A-15 to A-18)

5.0-34 In addition, the Staff compared Indian Point to other nuclear plants with regard to the potential for containment failure at the time of core meltdown from pressure spikes and reactor vessel missiles, and found that the Indian Point units were better than average at containing pressure spikes associated with reactor vessel melt through, and no more susceptible to reactor vessel missiles than any other plant. (Rowsome/Blond 5, pp. A-18 to A-19)

5.0-35 Finally, the Staff evaluated the effectiveness with which an initially intact containment can

bottle up core melt accidents both with and without the containment cooling and heat removal systems functional. The Indian Point containments appear to be among the more effective designs with regard to their capabilities to mitigate offsite radiological releases given a core melt. (Rowsome/Blond 5, p. A.-27; see Id., pp. A-20 to A-27) Additionally, since most of the accidents at Indian Point evolve slowly, there would be time for operator action to terminate the pressure rise altogether, thus further reducing the frequency of releases. (Licensees 1, p. 111) The Staff concluded that the Indian Point plants are well above average in their ability to withstand overpressurization failure scenarios. (Rowsome/Blond 5, p.A-15; see Id., pp. A-12 to A-18)

5.0-36 This advantage of the Indian Point containments over many other containments is due to the large containment volume in proportion to reactor lower level, the high failure pressure, and the use of basaltic concrete in the basemat. (Rowsome/Blond 5, A-25; see Id., pp. A-15 to A-26)

5.0-37 On the strength of the engineered safety features, containment design, and other Indian Point systems identified above, less than two percent of the inter-

nally initiated core melts lead to containment failure at Indian Point. (Paddleford, et al., p. 23)

B. RISK FROM INTERNALLY INITIATED EVENTS

5.0-38 The population density surrounding the Indian Point site has virtually no effect upon risks to individuals. Further, compliance issues surrounding emergency preparedness requirements, particularly the speed of an evacuation, do not appreciably influence individual risks. (Rowsome/Blond 5, p. B-3)

5.0-39 The evidence on the risk of early and latent fatality to individuals posed by internally initiated accidents, i.e., those triggered by spontaneous LOCAs, transients, transient-induced LOCAs and loss of offsite power, is quite clear. Indian Point Units 2 and 3 are among the safer plants licensed to operate by the Commission with respect to the risks to individuals posed by internally initiated severe reactor accidents. (Rowsome/Blond 5, p. B-4; Paddleford, et al., pp. 10-12)

5.0-40 The testimony indicates that the frequency of severe releases caused by internally initiated accidents is much lower at Indian Point than at any other

operating nuclear plants. This is confirmed not only by PRA bottom-line results, but by qualitative considerations and engineering judgment as well. The origin of this finding can be traced to (1) better than average severe accident prevention features in the original plant design, (2) the effects of the Director's Order of February 11, 1980, and (3) better than average severe accident mitigation. (Rowsome/Blond 5, p. B-4)

5.0-41 The only other factor beyond the comparative likelihood of severe releases and the comparative effectiveness of emergency response which affects a comparison of individual risks is the weather for the various sites. At the Indian Point site, the weather is entirely typical of the distribution of weather conditions among the various reactor sites as they affect the consequences of accidents. (Rowsome/Blond 5, pp. B-4 to B-5)

5.0-42 On the basis of Findings 5.0-38 to 5.0-41, the Indian Point units pose below average risks to individuals due to internally initiated events. (Rowsome/Blond 5, p. B-5)

5.0-43 Societal risks are proportional to the population density distribution around a given site. Up

to fifty miles, the population surrounding Indian Point is about ten times that of the median site. Zion and Limerick are quite similar. This puts the size of the population at risk near the top but not well above the spectrum of population surrounding other plants licensed by the Commission. (Rowsome/Blond 5, p. B-6)

5.0-44 The various comparisons of the results of other PRAs to the results of the IPPSS and Staff analysis of the Indian Point plants for internal initiating events show that the early and latent facility societal risks from those facilities are within the range of risks estimated for other nuclear power plants. (Paddleford, et al., pp. 14, 16)

5.0-45 The evidence has indicated that the early fatality risks 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 very low frequency interfacing systems LOCA event is essentially the only contributor to early fatality risk at Indian Point. (Paddleford, et al., pp. 24-25) Since the evidence with regard to source terms presented under Commission Question 1 shows a strong likelihood that there will be at least a factor of 10 reduc-

tion in the source terms for severe reactor accidents, (see Findings 1.1-276 to 1.1-312), few if any early fatalities would occur for any Indian Point accident scenario.

5.0-46 The better than average individual risk for internally initiated events compensates for the higher than average population and property values surrounding the Indian Point site. Therefore, the conclusion can be drawn that for severe accidents caused by these events, the societal risks posed by Indian Point are roughly average, i.e., well within the spectrum of risks posed by other plants. (Rowsome/Blond 5, p. B-6)

C. RISK FROM EXTERNALLY INITIATED EVENTS

5.0-47 While both individual and societal risks from external events are less clear because there is little basis for comparison among currently available PRAs, several conclusions can still be drawn based upon available evidence. (Rowsome/Blond 5, pp. B-5 to B-7)

5.0-48 Three recent plant changes made to Indian Point Unit 2 to reduce the effects of external events have been credited by the Staff in evaluating the risk posed by Indian Point Unit 2. These are:

1. Alterations to the Unit 2 control building to reduce the seismic fragility of the structure associated with interactions between the Unit 2 control building and the Unit 1 superheater building during earthquakes.

2. Alternate cables and breakers have been provided at Unit 2 to enable a component cooling water pump and a charging pump to be energized in the event that a fire in the cable tunnel or switchgear room disables the normal power supply to these systems.

3. The Unit 2 technical specifications were amended to mandate an anticipatory shutdown of Unit 2 should a hurricane threaten the site.

(Rowsome/Blond 5, p. C-3)

5.0-49 After these recent plant modifications, the Staff concluded that it is plausible that Indian Point poses an average to below average risk to individuals from externally initiated events. Although the Staff testified that this conclusion could be questioned due to lack of PRA comparability and large uncertainties surrounding external event risk assessment, the conclusion is strengthened by the fact that the Staff did not give any credit for alterations to the control room ceilings at the Indian Point units to reduce seismic fragility, nor did they eliminate conservatisms in estimating the frequency of hurricane-induced severe reactor accidents. (Rowsome/Blond 5, pp. B-5 to B-7, B-19) Further, uncontroverted testimony indicated that the recent reassessment of the seismic fragility of the containments of both units

results in the conclusion that the direct seismic failure of the containments is impossible. (See Finding 1.1-83) Since the Staff has given no credit for this reassessment, the likelihood that the Staff conclusion is correct is further increased. (See Finding 1.1-365)

5.0-50 There is no reason to believe that the risk to individuals in the vicinity of Indian Point from severe externally initiated reactor accidents lies above the spectrum of individual risks to persons similarly situated in the vicinity of other reactors since, even when the overall frequency of severe releases at Indian Point (after the plant modifications referred to in Finding 5.0-48) is compared with severe release frequencies from internal events only at other plants, Indian Point appears to be roughly average. (Rowsome/Blond 5, p. B-5) Further, for plants where externally initiated accident risk has not been analyzed, the overall risk from those facilities could only be higher if external events were to be analyzed and included. (Bley, Tr. 12678; Sholly, Tr. 12742; Rowsome, Tr. 12922)

5.0-51 The recent plant modifications identified in Finding 5.0-48 have substantially lowered the dominant externally initiated contributors to risk at Indian

Point. Therefore, the Staff concluded that it is less likely that Indian Point will pose a disproportionate share of the societal risk in the future than in the past. Once again, this conclusion is strengthened by the fact that the Staff analysis failed to take account of the Indian Point control room ceiling alterations, the containment fragility reassessment and conservatisms in the hurricane analysis, each of which can only further reduce the Indian Point risk. (Rowsome/Blond 5, pp. B-6 to B-7, B-9)

IV. COMPARISON WITH COMMISSION'S TASK FORCE RESULTS

5.0-52 One other overall risk comparison can be made. In June 1980, the Staff published NUREG 0715, "Report of the Task Force on Interim Operation of Indian Point." This report consisted of a brief PRA analysis which enabled a preliminary comparison to be made between Indian Point risks and those of other reactor sites and designs. The conclusion of the Task Force was that the Indian Point reactor design was about ten times better than the average reactor design, but that the site was about ten times worse than the average site. Therefore, the two factors offset each other, and Indian Point was not thought to pose undue public risk. The conclusions of

the Task Force were reaffirmed in the recent Staff testimony. (Rowsome/Blond 5, pp. A-3 to A-4)

5.0-53 Based upon the results of the IPPSS for internally initiated accidents, the early fatality risk curve lies more than an order of magnitude below the range of results presented in the Task Force study, employing the conservative source term assumptions used in IPPSS and the Staff testimony. A large part of this difference results from the fact that the Task Force did not 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. (Paddleford, et al., p. 18)

5.0-54 The Task Force had studied the effects on risk of both a typical pressurized water reactor (Surry, the Reactor Safety Study plant) at different sites and different plants at the same site (Indian Point). The IPPSS latent fatality risk curve lies within the range of the latent fatality risk curves calculated by the Task Force for the Indian Point site, and is below the range of curves calculated for the Surry reactor at various sites. (Paddleford, et al., p. 18)

V. SANDIA SITING STUDY

5.0-55 In its memorandum and order of November 15, 1982, the Board directed the parties to address the significance of the Sandia Siting Study, "Technical Guidance for Siting Criteria Development" (NUREG/CR-2239) to the comparison of risk among nuclear plants to which Commission Question 5 pertains. (November 15, 1982 Memorandum and Order, p. 19)

5.0-56 The purpose of the Siting Study was to provide for rulemaking purposes technical information relevant to the formulation and comparison of possible generic siting criteria for nuclear power plants. (Rowsome/Blond 5, p. A-29; Potter 5, p. 1) Using site-specific meteorology and population data (1970 census) and assuming an 1120 MWe reactor, NUREG/CR-2239 assessed the consequences of hypothetical severe power plant accidents. These consequences were calculated using an assumed "representative" set of generic siting source terms (SST) intended to cover the full spectrum of releases for severe light water reactor accidents. (Potter 5, p. 1; Rowsome/Blond 5, p. A-29; Sholly 5, pp. 6-7)

5.0-57 The Siting Study focused mainly on the three most serious generic source terms, SST-1, SST-2 and SST-3. (Potter 5, p. 1; Rowsome/Blond 5, p. 29) The report included "typical" probabilities which might be associated with each of the three assumed releases, but the absolute and relative probabilities associated with the accident source terms used in the study are generically based and are not a reflection of a specific design or a given facility. (Rowsome/Blond 5, p. 29) Therefore, the Study results are not to be used to evaluate questions of risk from existing plants as recognized and identified in the foreword to the report:

> There are very large uncertainties associated with these numbers. The absolute values and 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.

In order words, the uniform releases and associated frequencies of occurrence which were assumed for purposes of NUREG/CR-2239 should not be assumed to apply to real plants. Differences among real plants could affect risk estimates significantly. (Potter 5, p. 2) Thus, the picture presented in the Siting Study is a perspective on siting factors only, and the consequence estimates represent only the consequences of three hypothetical source terms which do not reflect any specific reactor design. (Rowsome/Blond 5, p. A-32)

5.0-58 The emergency response assumptions used for "best estimate" consequence modeling in the Siting Study is referred to as "summary evacuation." (Sholly 5, p. 7) The assumptions made for summary evacuation are that evacuation of people within a ten-mile radius takes place at a speed of ten miles per hour. In 30% of the evacuation simulations, the entire evacuee population was delayed for one hour; in 40% the population was delayed for three hours; and in 30% the population was delayed for five hours. (Potter, Tr. 12794-12795) No emergency response at all was assumed to take place beyond ten miles for 24 hours after plume passage. (Potter 5, p. 8; Potter, Tr. 12799-12800; Sholly, Tr. 12759)

5.0-59 When source terms other than the most serious SST-1 release were assumed in the Siting Study they were found not to result in significant differences between sites. When consequence calculations were done for the SST-2 and SST-3 source terms, the parties agreed that variations between sites in terms of early fatalities, early injuries and latent cancer fatalities were found not to be significant. (Rowsome/Blond 5, pp. A-30 to A-31; Potter 5, p. 3; Sholly, Tr. 12736)

5.0-60 UCS/NYPIRG asserts that the Sandia Siting Study provides a basis for risk comparisons between reactor sites because, assuming the occurrence of an SST-1 release and the best estimate emergency response assumptions, a comparison of Indian Point with the other ninety sites evaluated shows that the Indian Point consequence curves fall into the upper range of curves for all sites in each of the three consequences indices shown (early fatalities, early injuries, and latent cancer fatalities). (Sholly 5, pp. 7-13) The results of the study show that Indian Point Units 3 and 2 ranked sixth and seventh respectively in maximum calculated early fatalities. However, neither of the Indian Point fatalities appear in the first ten for maximum calculated latent fatalities. (Sholly 5, Appendix G)

5.0-61 The Staff testified that Sandia Siting Study results, assuming an SST-1 release and Summary Evacuation, show that Indian Point has higher mean early fatalities, early injuries and latent fatalities than the median site by factors of less than thirty, approximately fifteen and approximately six respectively. 'Rowsome/Blond 5, p. A-30) There is about a two and one half order of magnitude spread between the ninety-one sites evaluated for the probability of at least one early fatality and a three and

one half orders of magnitude spread in early fatalities at the one-in-a-thousand conditional probability level (Id., p. A-32) However, the testimony indicates that the absolute early fatality risk for many plants, including Indian Point, is very low. Absolute risks of this very low magnitude render differences between these low numbers for various plants and/or sites relatively unimportant. (Paddleford, et al., pp. 24-25) Further, there are no clear discontinuities in the family of curves. That is, there are no apparent break points which would indicate that one site or even a group of sites are out of the reasonable bounds of the family of curves. Therefore, none of the sites can be said to represent a unique extreme in the continuum of current sites. The curves for injuries and cancer fatalities further reinforce this notion, since the spread between sites is much smaller than for early fatalities. Thus, Indian Point is not unique or even unusual as far as the set of sites is concerned. (Rowsome/Blond 5, p. A-32)

5.0-62 The Siting Study's conditional risk curves for the ninety-one different sites show that latent cancer fatality risks does not vary greatly from site to site for a given release. (Potter 5, p. 13; Rowsome/Blond 5, p. A-32; Sholly 5, p. 8) This is because 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. For large areas, however, differences in population density among sites are relatively small. (Potter 5, pp. 13-15)

5.0-63 Therefore, the Siting Study shows that the Indian Point plants are close to the national average of the mean values for latent fatalities. Since the evidence indicates a lower frequency of containment failure for Indian Point than for many other plants, the narrow range of latent fatality consequences coupled with a lower frequency of containment failure support the conclusion that the Indian Point latent fatality risk is within the range of such risks posed by other nuclear power plants. (Paddleford, et al., pp. 25-26)

5.0-64 While the Siting Study offers the most comprehensive evaluation into the technical considerations of siting that has been performed to date, (Rowsome/Blond 5, p. A-31) there are several limitations inherent in the report which prevent its use in producing accurate risk assessments of existing nuclear power plants. (Potter 5, p. 2)

5.0-65 AS UCS/NYPIRG conceded on cross-examination, the Siting Study made no effort to model actual reactor-site combinations, its only variable being siting characteristics. There are very large variations in reactor design. These design differences can have a major impact on accident probabilities, so that overall risk may turn out to be higher at lower population sites than at Indian Point as more in depth PRAs are performed. (Sholly, Tr. 12738-12739, 12740-12741)

5.0-66 UCS/NYPIRG also acknowledged that the Siting Study is incomplete because it does not include probabilities for any of the releases modelled which are reactor design-specific, (Sholly, Tr. 12750), and that a complete view of the risk posed by a nuclear power plant requires an examination of both the site-specific probability of a release and the consequences to public health from that release. (Sholly, Tr. 12750, 12771)

5.0-67 The severe release which IPPSS found to be most probable at Indian Point, category 2RW, has no SST counterpart. Nonetheless, the frequency of the SST-1 release assumed in the Siting Study is the same as the frequency calculated in IPPSS for the delayed (and thus

less serious) 2RW release. The IPPSS release category which is the most similar to the Siting Study's generic source term SST-1, IPPSS release category 2, has a frequency of release at Indian Point which was determined to be much lower than the frequency assumed in the Siting Study for the SST-1 category. (Potter 5, p. 3) Because the Siting Study failed to include a release category for late containment failure, and because it assumed much higher frequencies for serious releases than those determined on a plant-specific basis, the use of the Siting Study results without adjustment for plant-specific considerations gives large overestimates of early fatality risk at Indian Point Unit 2. The assumption of early containment failure in place of late 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 prior to the commencement of evacuation. (Potter 5, p. 7)

5.0-68 It is also inappropriate to use the Siting Study SST-1 source term as a surrogate for IPPSS release category 2RW because the IPPSS analysis indicates that no early fatalities will occur beyond ten miles for release category 2RW even if there are are no emergency response actions taken for twenty-four hours after the

passage of the plume. (Potter 5, p. 8) The Staff agreed that when an evacuation and supportive medical treatment are implemented in response to this type of release, no early fatalities would occur. (Acharya Testimony, p. III.C-5) In this respect, IPPSS release category 2RW differs from IPPSS release category 2 and the Siting Study SST-1 because of somewhat lower release fractions and a substantially longer release duration for 2RW, modeled in IPPSS as a multi-puff release. This is another reason why the use of SST-1 risk estimates for the 2RW case result in substantial overestimates of early fatality risk. (Potter 5, p. 8)

5.0-69 The Siting Study's treatment of emergency response beyond ten miles also limits the value of using SST-1 risk estimates as a surrogate for IPPSS release category 2. As opposed to the Siting Study, which assumed no emergency response beyond ten miles, for IPPSS release category 2 it was assumed that ninety percent of the population from ten to fifty miles took shelter equivalent to that provided by the basement of a single family house, which results in a factor of 4 reduction in ground dose over that which would be received in normal cativities and a factor of about 9 reduction below what would be expected for outdoor exposure. The remaining ten percent of the

population was assumed in IPPSS to carry on normal activities, allowing for some failures in notification or in gaining access to shelter. For all persons, the exposure period was assumed to be twenty-four hours. (Potter 5, pp. 8-9) This dose assumption was made because such an emergency response would be relatively easy to implement and would be effective. It is inconceivable that such an emergency response would not be implemented in the face of significant doses affecting large numbers of people, although the Siting Study assumed to the contrary. (Id.) The authors of NUREG/CR-2239 acknowledge in the report that their assumption of no emergency response beyond ten miles may result in an overestimate of early fatalities. (Id., p. 2; NUREG/CR-2239, p. 2-51)

5.0-70 The failure to allow for any emergency response beyond ten miles in the Siting Study results in a factor of 5 increase in the maximum number of early fatalities calculated over the equivalent IPPSS results for release category 2. (Potter 5, pp. 9-10)

5.0-71 The warning to evacuate for an SST-1 release is assumed in the Siting Study to come one half hour before the release. (Potter, Tr. 12794) This one half hour warning time, when combined with the summary

evacuation delay time assumptions, (see Finding 5.0-58), means that no one in the ten mile EPZ begins to evacuate until at least one half hour after the release has begun, and many do not begin evacuating until four and one half hours after the release occurs. (Potter, Tr. 12788-12789, 12794-12795) Because the IPPSS 2RW release category results from a slowly developing overpressurization of the containment which takes at least twelve to twenty-four hours to evolve, and which affords substantial warning time for evacuation, evacuation in the Indian Point EPZ would begin for everyone prior to the occurrence of a radioactive release following a slow overpressurization failure. (See Findings 1.1-149, 1.1-249 to 1.1-251)

5.0-72 Based on the limitations of the Sandia Siting Study for purposes of modeling actual plant risk which are identified in Findings 5.0-65 to 50-71, the use of its 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. These limitations make it reasonable to conclude that the report is inapplicable for purposes of risk comparisons to other plant-site combinations as well. Thus the study results cannot be used alone to perform a comparative risk assessment for

existing nuclear plants. (Potter 5, p. 10; see Rowsome/Blond 5, pp. A-29, A-32)

5.0-73 Although the Siting Study 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 risks and on other questions raised in the course of this proceeding. (Potter 5, p. 11) One sensitivity study performed in the report shows the impact of different source term reductions on risk. The results of those studies are similar to those to which Licensee witnesses Stratton, Rodger and Potter testified under Commission Question 1. (See Findings 1.1-300 to 1.1-307) All risks decrease with source term reduction, but the decrease is more marked for early health effects. Large reductions effectively eliminate early fatality risk. (Potter 5, p. 11) In this regard, NUREG/CR-2239 explicitly recognizes that:

> Estimates of the number of early fatalities are very sensitive to source term magnitude. Mean early fatalities . . . are decreased dramatically (about two orders-of-magnitude) by a one order-of-magnitude decrease in source term SST1 . . . . Because the core melt accident source terms SST1-3 used in this study neglect or underestimate several depletion mechanisms, which may operate efficiently within the primary loop or the containment, consequence magnitudes calculated using these source terms may be significantly overestimated.

(NUREC/CR-2239, pp. 1-3) (emphasis in original)

5.0-74 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 risk reduction impact would be greatest for the highest population density sites. This would also result in lower and narrower ranges of absolute early fatality risk. (Potter 5, p. 11)

5.0-75 Sensitivity studies were also performed in the Siting Study utilizing different emergency response strategies. The findings of the sensitivity studies support and supplement the following conclusions:

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

2. 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 ten miles.

3. A strategy of evacuation within ten miles coupled with sheltering from ten to twenty-five miles is as effective as evacuation to twenty-five miles in reducing early fatality risks from the most severe releases.

4. Emergency response does not greatly affect latent fatality risk.

(Potter 5, pp. 12-13)

VI. ULTIMATE FINDINGS

5.0-76 The Board has heard evidence regarding comparisons of Indian Point risks with the risks posed by other nuclear power plants using several different methods of comparison, including the Commission's preliminary safety goals, other PRAs, the NUREG-0715 Task Force results, and comparisons made based upon engineering judgment. Although none of the comparisons are perfect, the evidence indicates, and the Board finds, that the totality of these comparisons is supportive of conclusions in both the Staff and Licensees' testimony that the Indian Point risk is low and within the range of risks of other plants. (Bley, Tr. 12679-12680)

5.0-77 The Board further finds that although the Sandia Siting Study represents a comprehensive review of considerations relating to power plant siting, its limitations prevent its results from being used alone to perform a comparative risk assessment for currently operating commercial nuclear power plants. However, the sensitivity studies presented in the report support conclusions which are consistent with the other comparisons made and results presented in these Findings of Fact.

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)	:	July 11, 1983

CON EDISON'S PROPOSED FINDINGS OF FACT REGARDING COMMISSION QUESTION 6

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COMMISSION QUESTION 6 AND CONTENTION 6.3

I. INTRODUCTION

6.0-1 Commission Question 6 asks:

> "What would be the energy, environmental, economic or other consequences of a shutdown of Indian Point Unit 2 and/or Unit 3?".

Contention 6.3, which is really a part of Question 6, states:

> "Considering the savings in operating expense which would result from shutting down Indian Point Units 2 and 3, and allowing for the ways in which cogeneration and conservation can mitigate the costs of replacement power, the net costs of shutdown are small; in fact, they are smaller than previous studies by UCS, GAO, or Rand suggest, and are entirely acceptable."

6.0-2 Testimony bearing on Question 6 and Contention 6.3 was presented by witnesses sponsored jointly by the Licensees as well as by witnesses sponsored separately by the Power Authority and by Con Edison; by witnesses sponsored by the Staff; by a witness sponsored by the New York State Energy Office; by a witness sponsored by UCS; by a witness sponsored by the Greater New York Council on Energy; and by witnesses sponsored by various members of the New York City Council.

6.0-3 All witnesses agreed that a shutdown of the Indian Point Units would increase the Licensees' cost of supplying electric service in future years and would require commensurate increases in the rates charged by the Licensees to their customers in Con Edison's service area. However, the estimates of the increases in rates which the ratepayers would have to pay as the result of a shutdown differed substantially, dependent primarily upon the witnesses' underlying assumptions as to the future prices of oil, the capacity factors projected for future operations of the Units, the projected operation and maintenance (O&M) expenses and capital charges associated with the Units, which would be avoided on a shutdown, the amount of energy which the Licensees' customers will take from them in the future, the number of years that the Units are likely to remain in operation, and the real discount rate used to determine the present value of a stream of future rate or cost increases. In addition, Licensees' estimates differed from the other estimates because it was the only one that reflected Statewide impacts as well as impacts in the Con Edison service area.

6.0-4 Higher future oil prices, higher capacity factors, higher energy consumption, longer

remaining lives for the Units, and a smaller real discount rate operate to increase the projected costs of a shutdown and the present value of those costs. Higher projected O&M expenses and capital charges operate to decrease the projected costs of a shutdown because the O&M expenses and capital charges represent costs that would be avoided on a shutdown, and would, therefore, be offsets to the production cost penalty.

II. THE DIRECT ECONOMIC CONSEQUENCES OF A SHUTDOWN

A. LICENSEES' ESTIMATES

6.0-5 Licensees' estimates of the cost burden which would be imposed on their ratepayers in the Con Edison service area and on the ratepayers throughout the State of New York as the direct result of a shutdown of Indian Point Units 2 and 3 were presented through the testimony of Eugene T. Meehan, of Energy Management Associates, Inc. (EMA), and Ms. Sally Hunt Streiter, of National Economic Research Associates, Inc. (NERA). Based on that testimony, Licensees estimated that the present value of the increased rates which ratepayers in the State of New York would have to pay over the next 25 years as the result of a shutdown of the Indian Point Units amounts to some $9.0 billion (Streiter Testimony,

Table 1.1); that $4.478 billion of that amount reflects the present value of rate increases, ranging from 33.3 to 22.7 percent, which would have to be paid by the Power Authority's customers in the Con Edison service area (Streiter Testimony, Tables 3 and 4); and that $3.362 billion of that amount reflects the present value of rate increases, ranging from 4.6 to 3.7 percent, which would have to be paid by Con Edison's customers. (Streiter Testimony, Tables 2 and 4) Licensees also estimated that the present value of the increased rates which the Licensees' ratepayers in the Con Edison service area would have to pay amounts to some $7.8 billion. (Streiter Testimony, Tables 2 and 3)

6.0-6 To arrive at these estimated cost impacts, Licensees first developed the production cost penalty of a shutdown, using EMA's Promod III System, a comprehensive tool for performing major production cost analyses, which is capable of simulating the operation of an integrated power pool in which transmission limitations affect the commitment and dispatch of the generating units operated by members of the pool. (Meehan Testimony, p. 3) The Promod III system was the only system used by any of the witnesses in this proceeding which was able to take account of transmission transfer

limits in the New York Power Pool and to simulate the Power Pool's dispatch on a statewide basis with and without the Indian Point Units. (Id.) The PROMOD III System also stands out from less sophisticated production costing programs in its use of an advanced probabilistic technique to properly consider the resultant impact of forced outages on fuel requirements, operating costs, and reliability. (Id., p. 4)

6.0-7 Licensees' estimated cost impacts reflect the following assumptions: Indian Point Units 2 and 3 will operate until their operating licenses expire in the years 2006 and 2009, respectively, at an average capacity factor of 63 percent to 2002, declining thereafter to 45 percent by the end of the license period (Streiter Testimony, Table 8); the price of oil, as the principal replacement fuel (Meehan Testimony, Table 4.7, p. 36), will, as forecast by ICF, Inc. for the New York Power Pool's 1983 Section 5-112 Report to the State Energy Office, decline by 6.7 percent in 1983 from the current world price of $34 a barrel, return to the real 1982 price by 1985, and thereafter increase at a real rate of 2 percent per year (Streiter Testimony, p. 6); energy consumption in New York State will increase by 1.4 percent per year, which will include a

growth of 1.3 percent per year in Con Edison's service area (Id., p. 19); O&M costs will be those shown in Licensees' detailed estimates of operation and maintenance expense through 1986 together with their estimates of required capital additions to the units to 1986, annualized over their remaining lives, and that after 1986, O&M expenses will increase at a real rate of one percent per year (Id., pp. 12-13); and the proper discount rate to determine the present value of these future cost increases is ten percent consisting of a real rate of three percent and an inflation rate of seven percent. (Id., p. 23)

6.0-8 Employing these assumptions as to the future capacity factors of the Indian Point Units, projected increases in oil prices, and future growth in energy consumption, together with the resource assumptions that were being relied on by the New York electric utilities in their preparation of the report of Member Electric Systems of the New York Power Pool pursuant to Section 5-112 of the Energy Law of New York State (Section 5-112 Report) (Meehan Testimony, p. 1), and the system dispatch predicted by the PROMOD III system, Mr. Meehan estimated that a shutdown would cause

the ratepayers in the State of New York a total production cost penalty of $18.0 billion over the 1984-1999 period, $16.0 billion of which would be borne by the ratepayers of Con Edison and the Power Authority in the Con Edison service area. (Meehan Testimony, p. 40) Mr. Meehan also estimated that the shutdown would result in the use of approximately 192 million additional barrels of oil in that 16-year period. (Id.)

6.0-9 Mr. Meehan conducted various sensitivity analyses to show the impact on his estimated production cost penalty of higher and lower fuel prices, higher and lower capacity factors, higher and lower load growth, and no conversion of oil plants to the use of coal (Meehan Testimony, p. 17, Tables 4.1, 4.2 and 4.3). All but one group of these sensitivity analyses (i.e., that relating to fuel prices) understate the economic impact of a shutdown because they include 1100MW of new coal capacity that was deferred indefinitely after Mr. Meehan completed his studies. (Meehan Testimony, p. 17)*

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* Only Mr. Meehan's base case estimate and alternate fuel cost sensitivity analysis were re-run to reflect the deferral of those generating units (Meehan Testimony, p. 17).

6.0-10 Mr. Meehan estimated that the production cost penalty of a shutdown would increase from $455 million in 1984 to $2.09 billion in 1999. (Meehan Testimony, Table 4.2, p. 31) Based on ICF's estimate that the real price of oil will increase at two percent per year after 1985, Ms. Streiter estimated that Mr. Meehan's production cost penalty will increase at a real growth rate of two percent per year from 1999 to 2006, when the operating license of Indian Point Unit 2 expires, and until 2009, when the operating license of Indian Point Unit 3 expires. (Streiter Testimony, p. 3)

6.0-11 The capacity factors for Indian Point Unit 2 and Unit 3 averaged only 53 percent and 51 percent, respectively, to the end of 1981 because both units experienced serious problems with major components which caused extended outages. (Streiter Testimony, p. 9) While Ms. Streiter recognized that her use of 63 percent as the capacity factor for the Licensees' base case therefore reflected some optimism, she pointed out that the lower capacity factors experienced by the Licensees were not the product of unidentified or random problems, but of known difficulties which have either been solved or are in the process of being solved. (Streiter, Tr. 13404-13406) In support of her optimism,

Ms. Streiter pointed out that Con Edison achieved a capacity factor of 62 percent in its last refueling cycle from May 1981 to January 1983. (Streiter Testimony, p. 11) She also pointed out that the nuclear power industry has made great efforts to improve its response time to technical problems through the sharing of data and through pre-outage technical analyses, and that the same reference case which reflects a capacity factor of 63 percent also reflects the Licensees' large expenditures on O&M and capital additions to the Indian Point Units. (Id.)

6.0-12 In response to the position she anticipated that some intervenors would take on the issue as to a reasonable capacity factor, Ms. Streiter testified that statistical data do not support the view that the capacity factors of large saltwater-cooled PWRs will decline steadily as they age.* She explained that the recent decline in capacity factors of saltwater-cooled PWRs, which has been labelled by some as an "age effect," is, in fact, due to recent steam generator

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* Dr. Richard A. Rosen presented a report by Energy Systems Research Group purporting to support this view (see pp. 6-19 to 6-21, infra).

problems at three plants (Streiter Testimony, p. 11), and that the data for one of those plants alone is statistically responsible for half the so-called "age effect" because it is the only saltwater-cooled PWR that is older than 10 years (Id., p. 11). Moreover, its extended outage, as Staff pointed out, had nothing to do with salt water, but involved seismic related improvements. (Feld, Tr. 14133)

6.0-13 While Ms. Streiter thought that 63 percent is "a likely and achievable number" (Streiter, Tr. 13414), she recognized that one could defend the use of 57 percent in the reference case. (Streiter, Tr. 13413-14) She testified that the use of a 57-percent capacity factor would reduce her present value from $9.0 billion to $7.583 billion. (Streiter, Tr. 13414).

6.0-14 Ms. Streiter also computed the present values of the rate increases indicated by Mr. Meehan's sensitivity analyses. With ICF's high estimate of oil prices, the present value was $10.8 billion; with ICF's low estimate of oil prices, the present value was $6.78 billion; and with low-low oil prices it was $5.88 billion. (Streiter, Tr. 13460) With a high capacity factor, the present value was $10.48 billion; with a low

capacity factor it was $7.58 billion (Id.). With a low O&M, the present value was $9.47 billion; with a high O&M, it was $8.5 billion (Id.). With a high load growth (2.1 percent), the present value was $10.85 billion; with low load growth (0.7 percent) it was $7.49 billion. (Streiter Testimony, p. 20; Streiter, Tr. 13460)

6.0-15 If the impacts of a high capacity factor, a high oil price, and a low O&M are aggregated, the present value goes to $12.86 billion (Con Ed. Exh. 14); if the impacts of lower capacity factors, low-low oil prices, and higher O&M expenses are aggregated, the present value goes to $3.3 billion (Id.). However, Mr. Meehan testified that it is highly unlikely that all the events that tend to either increase or decrease the penalty from that derived in the reference or base case would occur simultaneously. (Meehan Testimony, p. 38)

B. STAFF'S ESTIMATES

6.0-16 Staff's testimony on Commission Question 6 was presented by Messrs. William A. Buehring, Sidney A. Feld, Erastace N. Fields, Phillip R. Michelson, and Robert S. Wood.

6.0-17 Staff witnesses, using a computational approach rather than a cost production model (Buehring, et al. Testimony, p. 26), estimated a present value of $4.4 billion for Con Edison's service area alone (Id., p. 46). That estimate assumed a capacity factor of 50 percent (Id., pp. 30-31), no increase in the real price of oil (Id., p. 36), a replacement coal plant coming on line by 1990 (Id., p. 35), savings in nonrecurring items, O&M, and other avoided costs with a present value of $800 million (Id., p. 45), the operation of Units 2 and 3 to the years 2003 and 2006, respectively, and a real discount rate of five percent on top of an inflation rate of eight percent. (Id., p. 39) Staff noted that its estimate was sensitive to the real discount rate, and that, if a real rate of two percent, instead of five percent, were used, the present value of its gross production cost penalty would have been increased from $5.2 to $6.5 billion. (Id., p. 39) Ms. Streiter also showed the sensitivity of her estimates of present value to variations in the real discount rate. (Streiter Testimony, Table 20)

6.0-18 Staff witnesses testified that, while Staff typically assumes that nuclear plants will operate at an average capacity factor in the range of 50 to 70

percent, while the cumulative capacity factors for Indian Point Units 2 and 3 were 57.5 percent and 56.4 percent, respectively, through the end of 1980, while the GAO and the New York Power Pool have assumed a future capacity factor of 69 percent for these Units, and while 69 percent is an achievable performance level, but at the high end of the likely level, Staff was using 50 percent in this case to be "conservative". (Buehring, et al. Testimony, pp. 29-30)

6.0-19 Staff witness Feld testified that a comparison of the actual 1982 capacity factors for saltwater PWRs with the capacity factors which ESRG's regression model predicted for 1982 showed that the regression model almost always underestimated the capacity factor, that the error was not random, and that the model was biased in terms of underestimating capacity factors. (Buehring, et al., Tr. 14131-33) Staff counsel had previously gotten Dr. Rosen to agree that the regression model would be more accurate in its predictions for 1982 than in its predictions for years further out in the future. (Rosen, Tr. 13906)

6.0-20 Staff witnesses noted that the New York Power Pool's long-range estimates of fuel costs for

the period 1981 to 1991 showed an average annual compound rate of increase of about 12 percent in the price of high sulfur oil (Staff Testimony, p. 36), and that the price of oil has the potential for continued control by an international cartel. (Id.) However, for purposes of their estimate, they assumed that there would be no real increases in order to be conservative. (Id.)

6.0-21 Staff witness Feld agreed that Staff's present values were understated because a levelized fixed charge rate of 14.9 percent was used for the new coal plant beginning in 1990, that the actual fixed charge rate in the earlier years of the plant's life would have been significantly larger than 14.9 percent, and that the costs in the earlier years count more in the determination of present value than costs in the later years. (Buehring, et al., Tr. 14239-14243)

6.0-22 Staff also concluded that it was unlikely that the closing of the Indian Point Units would spur an increase in cogeneration or conservation (Buehring, et al. Testimony, p. 137), and that, even

with levels of cogeneration and conservation significantly greater than those likely to be implemented, the present worth of the replacement energy costs would be on the order of $4.0 billion if cogenerators sell electricity back to the Licensees at 75 percent of the Licensees' avoided costs. (Buehring, et al., Tr. 14234-35) Finally, Staff testified that if the sales back are made at 100 percent of avoided costs there would be no reduction in the cost impact of a shutdown because of cogeneration. (Buehring, et al., Tr. 14235)

C. INTERVENORS' ESTIMATES

6.0-23 On behalf of UCS, Dr. Vincent D. Taylor estimated that the present value of the net increase in costs which ratepayers would have to bear in the next 25 years if the Indian Point Units were shut down was $4.1 to $4.2 billion. (Taylor Testimony, p. 5) Dr. Taylor did not use a production cost model to determine the production cost penalty underlying this present value. Instead, he assumed that the generation from the Indian Point Units, operating at capacity factors of 55 percent (Unit 2) and 57 percent (Unit 3) (UCS Exh. II, addendum), would be replaced by oil-fired generation, using oil priced at the average U.S. retail

price for the first six months of 1980 (Id., p. 5), during the period 1981-87, and that the generation of the Indian Point Units would, for the balance of Dr. Taylor's 25-year costing period, be replaced by coal-fired plants having a total capacity of 1500 MW, costing $936 per kw. (Id., p. 7; Taylor Testimony, p. 17) Using a real discount rate of five percent, and nonrecurring savings of $375 million (UCS Exh. II, p. 7; Taylor Testimony, p. 29), Dr. Taylor arrived at his present value of $4.1 to $4.2 billion. (Taylor Testimony, p. 17)

6.0-24 This estimated cost impact would have been larger if, in evaluating the fixed charges for the new coal plant which he assumed would be built in the later years covered by his estimate, Dr. Taylor had not used a capital cost that was so much less per kw than the cost of coal plants now under construction. Dr. Taylor assumed that a new 1500 MW coal plant would be built for $936 per KW. (UCS, Exh.11, p. 7) When it was pointed out that the Somerset plant currently nearing completion was going to cost nearly twice that amount, Dr. Taylor suggested that his capital cost did not include interest capitalized. However, he ultimately withdrew that answer. (Taylor, Tr. 13305) Dr. Taylor then suggested that Somerset's cost of $1715 per KW

included inflation beyond 1979, whereas his figure was stated in 1979 dollars. However, that explanation ignored the fact that Somerset is scheduled for completion in 1984 (Meehan, Tr. 13184; Streiter Testimony, p. 22) and therefore could not reflect dollars beyond 1983. Inflation from 1979 to 1983 could not account for the difference between $936 per kw and $1715 per kw.

6.0-25 Mr. James M. Parmalee, appearing on behalf of the New York State Energy Office, used the General Electric Optimized Generation Planning Electric System Simulation Model to analyze the fuel and economic impacts of a shutdown on the ratepayers in Con Edison's service area. Assuming no real increase in the price of oil, a capacity factor of 57.7 percent, and an annual growth in energy consumption of 1.7 percent, Mr. Parmalee arrived at a present value of $2.3 billion for the increased production costs which he projected for the ratepayers in the Con Edison service area in the 15-year period 1982-1996. (Parmalee Testimony, p. 8) His conclusion was that "[t]here are likely to be significant economic and oil penalties associated with the shutdown of one or both of the Indian Point facilities, under various assumed capacity factors ranging from 47.1% to 68.3%". (Id., p. 9)

6.0-26 Mr. Parmalee agreed that recent changes in the plans for new generating units, as well as slippage in planned coal conversions, would increase the estimated production cost penalty. (Parmalee, Tr. 13765) He also agreed that the inability of his model to take account of transmission constraints among members of the Power Pool understates the penalty. (Parmalee, Tr. 13766) He assumed the same price for Canadian imports with Indian Point on or off, although a shutdown would increase the avoided cost of the Power Pool members and would therefore increase the price of Canadian power. (Parmalee, Tr. 13766; Streiter Testimony, p. 4) Finally, Mr. Parmalee ended his study with the year 1996, not because he thought the Indian Point plants will cease operating at that time, but because the data in the State Energy Master Plan which he used ran only to 1996. (Parmalee, Tr. 13768)

6.0-27 Appearing on behalf of various members of the New York City Council, Mr. David Schlissel and Mr. John Mavretich presented their estimate of the rate consequences of a shutdown without describing the mechanics of the model or method they used to estimate the production cost impact, other than to say that they compared the production costs

associated with the continued operation of the Indian Point Units with the production cost that would result from the generation of replacement power at oil-fired stations each year until 2009. (Schlissel and Mavretich Testimony, pp. 6-7) In that comparison, Messrs. Schlissel and Mavretich used capacity factors of 53.77 and 48.08 percent for Units 2 and 3, respectively (Id., pp. 8-9); historical O&M costs escalated at an inflation rate of seven percent (Id., pp. 11-12); the operating lives used by the Licensees (Id., p. 12); two estimates of oil prices, one which assumed that the price for oil in 1983 would be the same as in 1982 and thereafter escalate at the rate of inflation, and the other which assumed that the price of oil would decrease 15 percent between 1982 and 1983 and thereafter escalate at the rate of inflation (Id., pp. 15-16); and a real discount rate of four percent on top of an inflation rate of seven percent. (Id., p. 22) With those assumptions, they arrived at present values ranging from $2.9 to $4.7 billion, dependent upon the oil prices used. (Id., p. 23)

6.0-28 Dr. Richard A. Rosen, appearing on behalf of the Greater New York Council on Energy, presented a report by Energy Systems Research Group (ESRG) which estimated that the present value of 15

years of net increases in cost attributable to a shutdown was $746 million. In his testimony, Dr. Rosen stated that the recent drop in oil prices would practically wipe out that penalty (Rosen Testimony, pp. 6-7), although he did not modify ESRG's estimate. (Id., p. 5)

6.0-29 There are several reasons why ESRG's estimated present value is so much smaller than any other estimate in this proceeding. The principal of these reasons is that ESRG assumed that the capacity factors of the Units would equal their historic averages in the year 1982 and thereafter steadily decline. (ESRG Report, p. 29) Figure 1, page 27, of the ESRG report shows that, for Indian Point Unit 2, ESRG starts with a capacity factor of 55 percent in 1982, and reduces it steadily to 30.8 percent in the year 2000, producing an average of 42.9 percent for that period. Figure 2, on page 28, shows that, for Indian Point Unit 3, ESRG starts with a capacity factor of 53 percent in 1982, and reduces it steadily to 32.5 percent in the year 2000, producing an average for that unit of 42.8 percent. Given Ms. Streiter's testimony that a change of six percentage points in the capacity factor occasions a change of about $1.5 billion in her present value of the

projected cost increases (Streiter, Tr. 13416), it is apparent why ESRG's present value is so much less than that presented by any other witness.

6.0-30 Dr. Rosen claimed that his conclusions with respect to the future capacity factors of the Indian Point Units were supported by the testimony of Dr. Louis J. Perl before the Pennsylvania Public Utility Commission in 1981. (Rosen, Tr. 13842) However, Dr. Rosen agreed that it was Dr. Perl's view that capacity factors would be 50 percent at the outset of a unit's operations, increase to about 70 percent after ten years of operation, remain at 70 percent for the next ten years, and then decline to 20 percent at the end of the unit's book life. (Rosen, Tr. 13842-43) Asked why he was willing to accept Dr. Perl's estimated end points but not his intermediate points, Dr. Rosen stated that there was no operating data to support his intermediate points (Rosen, Tr. 13845). ESRG also purported to find support for its conclusions as to the future capacity factors in a multivariate regression analysis of capacity factors as measured by several explanatory variables. (ESRG Report, p. 25) However, Dr. Rosen testified that there was no operating data for the age bracket his regression analysis was purporting to

predict (Rosen, Tr. 13851), and that he was relying on operating data with respect to plants that were "six or seven to perhaps ten or twelve" years old. (Rosen, Tr. 13851).

6.0-31 Another factor contributing to the relatively small penalty derived by ESRG was its assumption -- reached without any analysis of the nature or cause of the large increases in O&M expenses recently experienced by Con Edison and the Power Authority (Rosen, Tr. 13867-68, 13872-73), and without any attempt to determine whether those unprecedented increases were likely to be repeated (Rosen, Tr. 13872-73) -- that O&M expenses could be projected for the Licensees by applying a simple linear least squares fit to their experienced costs beginning with the year 1981 to arrive at O&M costs (including purchased power) in 1997 which are more than 6.5 times the O&M costs in 1983, and which amount to $107.3 million in 1981 dollars compared to Con Edison's estimated 1983 O&M costs for the station of about $41 million. (ESRG Report, p. 44; Table 5, p. 32) While ESRG was willing to project O&M expenses in this manner, Dr. Rosen opined, without explanation, that the same technique would be inappropriate for forecasting oil prices. (Rosen, Tr. 13874)

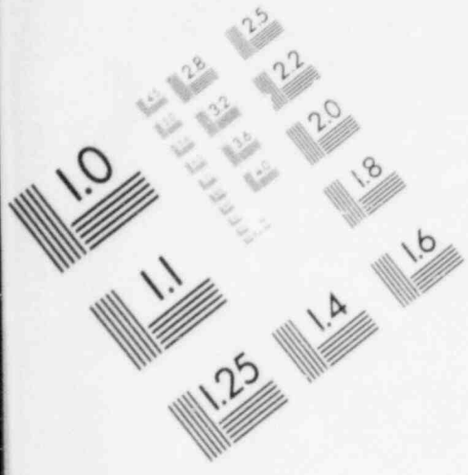
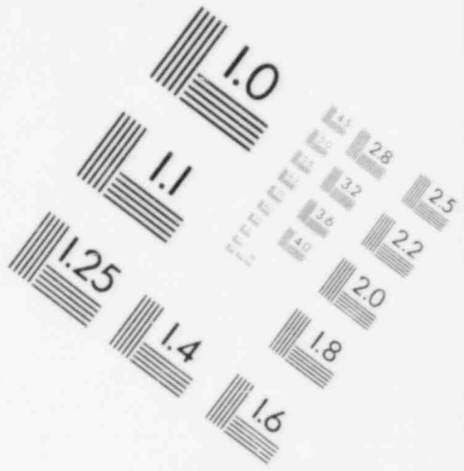
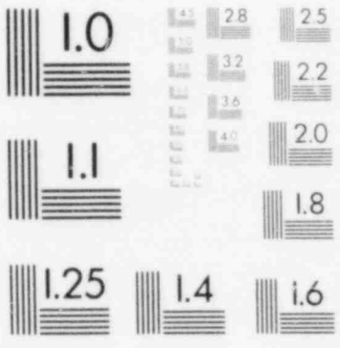
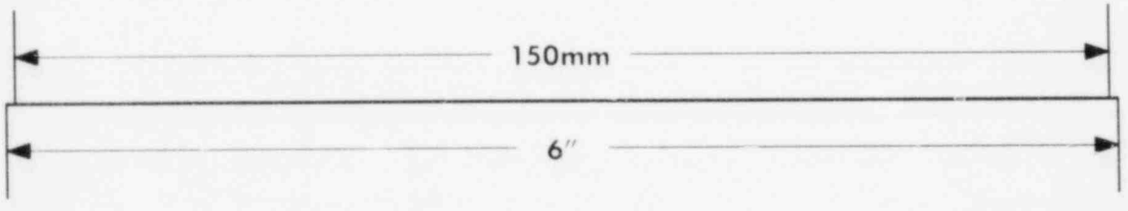
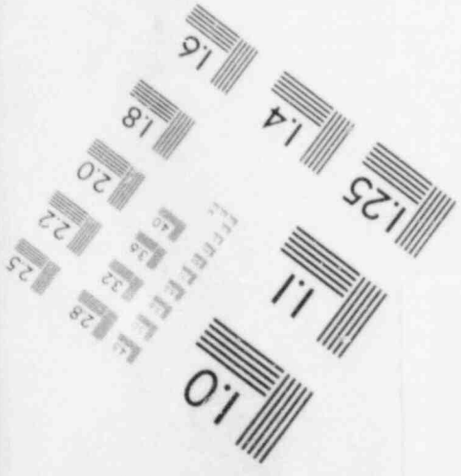


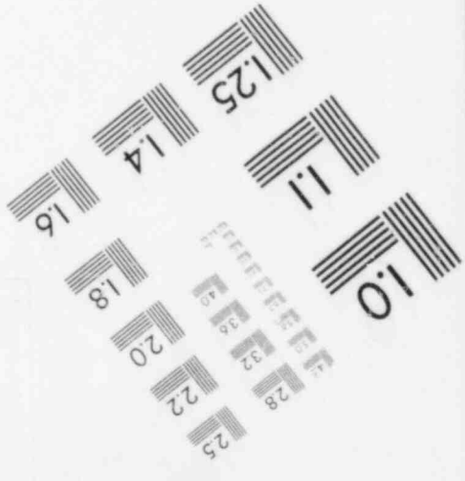
IMAGE EVALUATION TEST TARGET (MT-3)











6.0-32 Ms. Streiter undertook the analysis of the Licensees' O&M costs which ESRG did not make. As a result of that analysis, she determined that there were three major aspects to the sharp recent increases over the 1972-77 levels: first, there were one-time increases in basic staffing levels, including guards, training and operating personnel, and in security and other contracts, pursuant to various regulatory requirements, which are expected to continue, but not grow in the future; second, there was a series of one-time expenditures also related to regulatory requirements, which are expected to decline gradually; and, third, recent expenditures related to specific operating problems, particularly the sleeving program at Indian Point 3, are projected to be one-time expenditures. (Streiter Testimony, p. 14)

6.0-33 Still another factor contributing to the substantial difference between the penalty estimated by ESRG and the penalties estimated by the witnesses for other intervenors is ESRG's assumption that the Kwhr consumption in Con Edison's service area will decrease from 37.4 billion Kwhr in 1983 to 35.8 billion Kwhr in 1988 -- a five-year reduction of 4.2 percent (Rosen, Tr. 13828-29) and that consumption over the entire 15-year

period would decrease from 37.4 billion Kwhr in 1983 (ESRG Report, Appendix F, p. 31) to 37.1 billion Kwhr in 1997. (Id., p. 87) ESRG made no attempt to square that assumption with the fact that sales in the Con Edison service area increased by 4.1 percent in the five-year period 1977 to 1982. (Rosen, Tr. 13830) Neither did it attempt to explain why the New York utilities, including Con Edison, would project growth in sales*, if they did not believe there will be growth, given that rate regulation in New York is based on a projected test period and that a projected increase in sales operates to reduce the amount of rate relief allowed by the Public Service Commission. While the ESRG Report refers to a conservation scenario that was developed in a June 1981 study, the scenario described and the study were not introduced into evidence. (ESRG Report, p. 18) Finally, the assumption of no growth (or a slight decrease) in consumption in ESRG's mid-range case is not supported by that June 1981 study because that study shows a growth of 0.5 percent for its mid-range case. (Testimony of George L. Fitzpatrick, pp. 28-29)

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* As Mr. Meehan pointed out, the data he used showed that the Power Pool was forecasting a growth in energy consumption of 1.4 percent per year between 1984 and 1999. (Meehan Testimony, Table 3.1, p. 14)

6.0-34 Finally, although the conversion of Ravenswood 1 and 2 to coal would be economical whether or not Indian Point is in service (Rosen, Tr. 13803), ESRG didn't include the conversion in its base case with Indian Point in service. Dr. Rosen agreed that, if ESRG had included the conversions in its base case with Indian Point in service, its penalty would have been larger (Rosen, Tr. 13809, 13881-82), that the increase in the penalty would have been $83 million in 1991 (the difference between $300 million of fuel savings attributable to the conversion to coal and $217 million attributable to the capital and incremental fixed O&M costs of the coal conversion) (Rosen, Tr. 13803-13804), and that the increase would grow over the years (Rosen, Tr. 13805). All this led Staff Counsel to inquire whether Table 1 of the ESRG Report which is entitled Required Revenue Impact of Indian Point Retirements, shouldn't more correctly have been entitled Required Revenue Impact of Indian Point Retirements and Impacts of Coal Conversion at Ravenswood. (Rosen, Tr. 13901)

D. TESTIMONY ON COGENERATION AND CONSERVATION

6.0-35 Appearing on behalf of various members of the New York City Council, Dr. Barry Commoner, and Mr. Richard Schrader testified that, in lieu of operating Indian Point, Con Edison and the Power Authority should institute a program of conservation, by inducing the accelerated replacement of present appliances, such as refrigerators, air conditioners, and light bulbs, with more energy-efficient appliances. (Commoner and Schrader Testimony, p. 1)

6.0-36 Messrs. Commoner and Schrader agreed that their references to conservation did not take account of conservation already forecast by Con Edison and the Power Authority (Commoner and Schrader, Tr. 13955); that, under their proposal, people who don't replace their appliances would end up paying for those who do (Common and Schrader, Tr. 13978); that, while their proposal would call for the Power Authority, as well as Con Edison, to initially finance the purchase of new appliances, they didn't consider the question whether the Power Authority could do that and maintain the tax exempt status of its bonds under the Internal Revenue Code (Commoner and Schrader, Tr. 13996-97); that

they had not really considered where the billions of dollars necessary to purchase the new appliances would come from, or what the annual cost to the ratepayers would be if they had to pay a return on the capital required to finance the new appliances. (Commoner and Schrader, Tr. 14015, 14025)*

6.0-37 Mr. George L. Fitzpatrick testified on behalf of the Licensees that the members of the New York Power Pool have accounted adequately for conservation potential in the long-range forecasts they submitted in the State Energy Master Plan II (SEMP II) hearings (Fitzpatrick Testimony, pp. 6-10). Mr. Fitzpatrick also testified that the energy consumption reflected in Licensees' base case reflects the potential for conservation (including conservation achievable through cost-justified subsidies) in the Con Edison service area and may even overstate that potential. (Id., pp. 18-27) And he pointed out that the Executive Summary of SEMP II pointed out that its forecasts

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* Dr. Commoner stated that he wouldn't be concerned if Con Edison's stockholders had to absorb the cost of financing these appliances out of profits. (Commoner and Schrader, Tr. 14018, 14021)

reflected significant conservation resulting from rising energy prices, mandated efficiency standards and state conservation programs. (Id., p. 11)

6.0-38 On the subject of cogeneration, Dr. Commoner testified that he didn't know that the Totem on-site generators he had referred to in his direct testimony would have to be operated in parallel with the Company's system (Commoner and Schrader, Tr. 14030), and that every Totem connected to Con Edison's secondary network would, therefore, require an expensive network protector to assure that there was never any feedback of energy into the secondary system. (Commoner and Schrader, Tr. 14033) Dr. Commoner agreed that there would be environmental problems associated with any large proliferation of natural gas-fired cogeneration. (Commoner and Schrader, Tr. 14035)

6.0-39 Appearing on behalf of the Licensees, William J. Wagers explained why he didn't believe that natural gas-fired cogeneration was a likely candidate to replace the output of the Indian Point units. Mr. Wagers testified that the spread of such cogeneration facilities would be limited by physical constraints, such as the air pollution problems and natural gas

supply problems which would occur on any large-scale conversion. (Wagers Testimony, p. 2) Mr. Wagers also testified that Con Edison's most recent analysis, made in July 1982, indicated that only 72 customers, with a coincident peak load of 296 MW could, on economic grounds, have the incentive to become cogenerators (Id., p. 5); that in Public Service Commission Case 27574, which was a proceeding to examine all aspects of cogeneration, the Commission's Staff concluded that the "potential exposure to a loss of customers to on-site generation is low", and consisted of less than 25 customers with a coincident peak load of less than 100 MW (Id., p. 5); that the State Energy Office in its State Energy Master Plan, issued March 1982, estimated the potential for cogeneration facilities in the New York City area to be 380 MW, of which only 200 MW were likely to come on line by 1996 (Id., p. 7); and that the actual conversions to cogeneration have been much lower than Con Edison has estimated. (Id., p. 9)

6.0-40 Mr. Allen M. Stewart testified on behalf of the Licensees that 50 billion cubic feet of natural gas per year would be required to fuel 1,000 MW of gas-fired cogeneration (Stewart Testimony, p. 2); that that is 26 percent of the total quantity of gas

purchased by Con Edison in 1982, and 16 percent of the gas requirements for both Con Edison and Brooklyn Union in the same year (Id.); and that the availability of such an additional supply of gas would be "very uncertain, to say the least". (Id., p. 4)

6.0-41 Dr. Peter C. Freudenthal testified on behalf of Con Edison that his in-depth analyses of the environmental consequences of on-site generation in New York City had led him to conclude that the use of internal combustion engines with rooftop chimneys would be of particular concern with respect to oxides of nitrogen (NO_x) emissions, regardless of whether the engines are oil- or gas-fired. (Freudenthal Testimony, pp. 2-3) A proliferation of these types of fuel-burning equipment in an urban area -- Dr. Freudenthal stated -- would cause ambient nitrogen dioxide (NO_2) levels to rise and might cause the national ambient air quality standard (NAAQS) for NO_2 to be exceeded. (Id., p. 3)

6.0-42 Dr. Freudenthal testified that the nitrogen oxide emissions would be much less if boilers were used for the cogeneration instead of the internal combustion engines envisioned by Messrs. Commoner and Schrader (Freudenthal, Tr. 13695). However, Dr.

Freudenthal pointed out that there would still be a dispersion problem and an air pollution problem -- albeit a less severe problem than would be occasioned by the sort of cogeneration that Messrs. Commoner and Schrader were talking about and that Dr. Freudenthal was responding to in his testimony. (Freudenthal, Tr. 13696-97)

E. TESTIMONY ON PRICE ELASTICITY

6.0-43 Appearing on behalf of the Licensees, Dr. George C.S. Wang described the econometric model that the Company uses to forecast electric sendout in its electric rate cases, and testified that the short-term and long-term price elasticities presented by the Company in its most recent electric rate case were -.10 and -.25, respectively. (Wang Testimony, p. 3) Dr. Wang also testified that the price elasticity of -.4 referred to in ESRG's Report was a gross overstatement of the price elasticity of demand in Con Edison's service area. (Id., p. 2) Finally, Dr. Wang pointed out that, if Con Edison had used a short-term price elasticity of -.4, instead of -.1, in its most recent electric rate case, its revenue requirement would have been increased by $270 million. (Id., p. 4)

F. TESTIMONY ON OIL DEMAND, SUPPLY AND PRICE

6.0-44 Mr. Melvin A. Conant, appearing for the Power Authority, testified that while some analysts believe that further declines in the price of oil are possible, moderate economic recovery should restore stability and strength to oil prices as well as increase demand for oil. (Conant Testimony, p. 20) Mr. Conant also testified that measures which lead to increased oil demand, and thus to greater amounts of oil imports, are contrary to the national interests of the United States. In support of these two conclusions, Mr. Conant made the following points:

(a) Additions to oil reserves are not matching the amount of oil being pumped and every producer of oil will keep in mind the fact that oil is a diminishing resource which will eventually be exhausted (Id., p. 65);

(b) Increased requirements for oil by the less developed countries and by Communist nations, rather than increases in consumption by the industrial nations of the free world, will likely be the cause of increased oil consumption over the next 15 years (Id., p. 15); and

(c) By the end of this decade, there could be a greater volumetric requirement for oil imports than OPEC has ever exported. If, in the nineties, very substantial additional recoverable resources are not available, price increases of very substantial amounts can be expected as nations compete for their needs. (Id., p. 19)

6.0-45 Appendix 1 to Ms. Streiter's testimony shows that ICF's 1985 forecasts of oil prices are not out of line with the 1985 forecasts of other analysts and that its 1990 forecast is below the 1990 forecasts of those analysts. Ms. Streiter also pointed out that, while there are analysts who now think oil prices have levelled off, the same attitude prevailed in 1978 -- after four years of stable prices -- and that prices doubled in the following year. (Streiter Testimony, p. 8)

G. ULTIMATE AND SUBSIDIARY FINDINGS OF FACT

6.0-46 Given the several significant assumptions which underlie and affect any estimate of the net cost of a shutdown, and the length of the period over which we must predict costs as well as operating

results, a finding as to the dollar cost of a shutdown must be heavily judgmental. We emphasize, however, that the lowest reasonable estimate is still a very large number.

6.0-47 With respect to that part of Question 6 which addresses economic consequences, we find that a permanent shutdown of the Indian Point Units would result in substantial future rate increases to the ratepayers in the State of New York, and most particularly to the ratepayers in Con Edison's service area, over the remaining service lives of the Units, and that the present value of those ongoing future rate increases is likely to range between $7.6 billion and $9.7 billion. Given that rate increases of that magnitude would constitute a serious economic burden to any group of ratepayers -- let alone a group which includes ratepayers who are already paying the highest electric rates in the country -- this finding perforce mandates our rejection of Contention 6.3. In support of these ultimate findings, we make the following findings:

(a) Of the various production cost penalty modeling programs set forth in the parties' testimony, the PROMOD III System is the most appropriate program to

use in developing an estimate of the production cost penalty because it is the only program which takes account of transmission constraints in the State, provides a projection of forced outages and a Power Pool dispatch with and without Indian Point, and makes it possible to estimate the production cost penalty for the State as a whole as well as for Con Edison's service area. (Meehan Testimony, pp. 3-4)

(b) Ms. Streiter presented a persuasive case for the use of a capacity factor of 63 percent (Streiter Testimony, pp. 9-12), but also recognized that a case could be made for the use of 57 percent. (Streiter, Tr. 13413-14) In light of that testimony, we think that the capacity factor can most appropriately be stated in terms of its probable range, which we find to be 57 to 63 percent. On the basis of Ms. Streiter's calculation (Streiter, Tr. 13414), the use of a 57-percent capacity factor reduces the present value of ongoing future rate increases attributable to a shutdown of the Indian Point units from $9.0 billion to approximately $7.6 billion.

(c) ICF, Inc., the organization which forecasts oil prices for purposes of long-range planning by the New York Power Pool, forecasts that the price of oil

will decline by 6.7 percent in 1983, return to the 1982 price level by 1985, and thereafter increase at a real rate of two percent per year. (Streiter Testimony, p. 6) That forecast is reasonable. However, if we rejected that forecast, and concluded that there will be no increase in the real price of oil over the rather long period we are looking at,* the low end of our range would be reduced by some $2.2 billion, or from $7.6 billion to $5.4 billion* and that would still be a very large cost penalty.

(d) Based on the testimony before us, a reasonable annual rate of growth in the statewide consumption of energy, which adequately reflects the potential for conservation, is between the 1.4 percent used by the Licensees and the 1.7 percent used by the State Energy Office. The 1.4 percent is the growth rate which the members of the New York Power Pool were using in the preparation of their Section 5-112 Report for 1983 when Mr. Meehan made his study. The 1.7 percent is the growth rate used in the 15-year forecast of energy

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* Apart from ESRG, which tells us that the oil prices assumed in its mid-range case are somewhat higher than Licensees' oil prices (ESRG Report, p. 79, fn 14), the other parties assumed that there would be no real increase in the price of oil.

requirements in the SEMP II. A growth rate of 1.7 percent produces a present value that is some $700 million greater than the present value produced with a growth rate of 1.4 percent.

(e) For projected O&M, the Licensees used their detailed estimates of costs through 1986 together with their estimates of required capital additions to 1986, annualized over the remaining lives of the units. After 1986, Licensees estimate that O&M will increase at a real rate of one percent. (Streiter Testimony, pp. 12-13) These O&M expenses, which are substantially greater than the average O&M per kw being projected for other nuclear plants in the United States (Streiter Testimony, p. 13, Table 11), are reasonable.

(f) Licensees' assumption that the Indian Point Units will continue to operate until the expiration of their licenses is reasonable, and that the economic cost of a shutdown should therefore be determined for the period 1984-2006, in the case of

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* Ms. Streiter testified that her present value of $9.0 billion would be reduced to $6.78 billion with the use of ICF's low estimate of oil prices. (Streiter, Tr. 13460).

Indian Point Unit 2, and for the period 1984-2009, in the case of Indian Point Unit 3, is reasonable. (Streiter Testimony, p. 1)

(g) Licensees' use of a long-term real discount rate of three percent to determine the present value of the future rate increases which would be required by a shutdown is reasonable. (Streiter Testimony, p. 23)

(h) There are other less important assumptions which enter into an estimate of the production cost penalty occasioned by a permanent shutdown, i.e., the dates on which existing oil-fired plants may be converted to coal (Meehan Testimony, Table 3.4, p. 18), the amount of energy which may be imported from Hydro Quebec and Ontario Hydro and the prices of that energy (Meehan Testimony, pp. 22-25), and the cost of decommissioning the Indian Point Units. (Streiter Testimony, pp. 23-25) To the extent that the treatment of these other and less important assumptions in Licensees' estimates is different from their treatment in any of the other estimates, the Licensees' position on each of them is reasonable.

(i) The possibility of cogeneration does not constitute an offset to the economic cost of a shutdown.

III. THE INDIRECT ECONOMIC COSTS OF A SHUTDOWN

A. LICENSEES' TESTIMONY

6.0-48 Dr. Frederick C. Dunbar testified on behalf of the Licensees that the Metropolitan Transportation Authority (MTA), which is New York City's largest power consumer, will be particularly hard hit by a shutdown and the rate increases which would result from it. (Dunbar Testimony, p. 1) According to Dr. Dunbar, MTA's power costs, which are a major component of the MTA's operating costs, would increase by 29.5 percent, and the total operating costs would increase by 2.2 percent as a result of a shutdown. (Id., pp. 1-2)

6.0-49 Dr. Dunbar testified that, in response to increased power costs, the MTA will have to increase fares and/or reduce service (Id., p. 1) because it is unlikely it could get a subsidy to offset such an increase. (Dunbar, Tr. 13617) In fact, federal operating subsidies are expected to decline, if not disappear altogether, under federal budget cuts (Dunbar

Testimony, p. 6), and the battle in the New York Legislature to save the 75-cent fare was a long and bitter one, which would not bode well for any subsidy to offset an increase in power costs. (Id., p. 7)

6.0-50 Dr. Dunbar explained that the size of any fare increase effected by the MTA to pay for its higher electricity bills would depend on the additional revenue required to cover the increased cost of power and the additional revenue to compensate MTA for riders which are lost as a result of increased prices. An increase of 5.03 percent in fare box revenues of $870 million would be necessary to cover increased power costs of $43.8 million. (Dunbar Testimony, p. 12) Using an average long-run fare elasticity of demand for the transit system, equal to -.371, which Dr. Dunbar derived with an econometric model (Id., pp. 10-11), Dr. Dunbar calculated that a total fare increase of 8.0 percent would be required to produce an increase in revenues of 5.03 percent (Id., p. 12). With the current fare of 75 cents, the fare would therefore have to increase by 6 cents a ride to cover the higher cost of electricity. (Id., p. 13)

6.0-51 Dr. Dunbar also presented estimates, based on regression analyses having transit fare as one of the variables affecting employment, which show that each percent increase in transit fares causes New York City employment to decline .04 percent (Id., p. 14). An increase of 8.0 percent in the fare would therefore produce a decline of 0.32 percent in New York City employment. (Id.) With a predicted employment of 3,548,500 in New York City in 1986, Dr. Dunbar concluded that 11,400 jobs would be lost. (Id.)

6.0-52 Dr. Dunbar estimated that the decline in New York City aggregate personal income and tax revenues would be proportional to the decline in employment. Using 1986 as the forecast year, income would decline by $323 million, and City Tax revenues would decline by $34 million. (Id., p. 15)

6.0-53 Dr. Dunbar pointed out that the MTA might try to cut service in lieu of increasing fares. However, a reduction in service, Dr. Dunbar testified, would have a greater impact on ridership than an increase in fares because transit travel demand is known to be more responsive to changes in trip time than to changes in price. (Id.) Using studies available from

the New York State Metropolitan Transportation Council, Dr. Dunbar determined how a one-percent decline in funds available for transit operations would affect the average travel time, then determined the decline in transit travel from the increase in travel time, and then assumed that lost transit trips are related to New York City employment as if they had been caused by an increase in fare. (Id., pp. 15-16) Dr. Dunbar estimated that the decrease in quality of service would cause a 3.96-percent decline in ridership. Given that a 2.97-percent decline in transit riders equated to a 0.32-percent decline in employment, in the case of a fare increase, Dr. Dunbar estimated that the 3.96 percent decline in transit riders occasioned by reduced service would equate to an 0.43-percent decline in employment. (Id., p. 20) This gave him an estimate of 15,300 lost jobs, a loss of $434 million in personal income, and a loss of $46 million in New York City tax revenues. (Id.)

6.0-54 Drs. Charles R. Dean, Paul H. Rubin, and Harold M. Hochman testified on behalf of the Power Authority that the cost of a shutdown would not be distributed evenly among the ratepayers in Con Edison's service area. The major identifiable group which would

lose more than the average would be the subway and bus riders, who tend disproportionately to be relatively low income persons (Dean, Rubin and Hochman Testimony, p. 26). The economy of the area, Messrs. Dean, Rubin and Hochman testified, may also suffer losses beyond those testified to by Meehan, Streiter and Dunbar if electricity users in the Con Edison service area reduce their consumption of goods and services other than electricity, and those reductions are reflected in a decrease in economic activity. (Id., p. 27)

6.0-55 Ms. Streiter testified that a shutdown would require a sharp rate increase for the Power Authority's Southeast New York customers. To the extent, however, that such increases would raise the Power Authority's rate levels above those of an alternate supplier, the Power Authority's downstate customers might well seek another supplier. This would threaten the Power Authority's ability to meet required payments under its bond resolution and would thereby have a substantial adverse impact on the Power Authority, its future projects and other governmental entities in New York State. Accordingly, the Power Authority expects that the embedded costs of Indian Point Unit 3 might have to be recovered from the customers of the Power Authority's remaining revenue-

producing facilities in proportion to the ability of each of those facilities to carry such costs and still produce power at competitive rates. (Streiter Testimony, p. 29)

6.0-56 Although the Gov. Charles Poletti plant, which is one of the two Power Authority plants serving its customers in the Con Edison service area, would be one of the Power Authority's remaining revenue-producing facilities, the Power Authority thinks it quite possible that none of the costs of closing Indian Point could be recovered from ratepayers served from the Poletti plant because, on a stand-alone basis, it estimated that Poletti rates would just barely be competitive with Con Edison's rates. Accordingly, the Power Authority customers asked to bear the cost of closing Indian Point might be those located in Upstate New York who would be customers of hydroelectric projects. (Streiter Testimony, pp. 29-30) The Power Authority estimated that the hydroelectric project rates would have to be increased immediately by about $70 million per year, or by about 45 percent above the current rates paid by customers of the hydroelectric projects. (Id., p. 30)

B. ULTIMATE AND SUBSIDIARY FINDINGS OF FACT

6.0-57 We find that there will be indirect economic costs attendant upon a shutdown because the MTA, which is New York City's largest power consumer, will be hit particularly hard by the rate increases which would follow a shutdown. Such rate increases would have to be offset by a subsidy, or an MTA fare increase, or a cut in MTA's services. Given the improbability of an additional subsidy, MTA would have to raise its fares or cut its services, either of which would cause a loss in jobs, a loss in personal income, and a loss in tax revenues in New York City. (Dunbar Testimony, pp. 13-20) Moreover, the economy of the area could suffer further losses if electricity users in the Con Edison service area reduce their consumption of goods and service other than electricity. (Dean, Rubin and Hochman Testimony, p. 27)

6.0-58 We also find that the Power Authority, because of competitive rate conditions, may not be able to recover the costs of a shutdown from its customers in Con Edison's service area, and would, instead, have to impose those increases, amounting to $70 million or about 45 percent per year, on its hydroelectric

customers in Upstate New York. (Streiter Testimony, pp. 29-30)

IV. THE ENVIRONMENTAL AND REGIONAL SOCIOECONOMIC CONSEQUENCES OF A SHUTDOWN

A. STAFF'S TESTIMONY

6.0-59 Appearing on behalf of Staff, a panel of witnesses chaired by Mr. Charles W. Billups, an Aquatic Scientist in the Commission's Office of Nuclear Reactor Regulation, testified that with the continued operation of Indian Point no irreversible impacts on aquatic resources will occur over the ten-year term of the settlement agreement reached in the EPA's Hudson River Power Case. (Billups' Panel Testimony, pp. 8-9) If the units were shut down and replaced by existing oil-fired units, the aquatic impacts would be dispersed over several aquatic systems. (Id., pp. 9-10) If the units were replaced by coal units, the aquatic impact would depend upon the type of cooling used. (Id.)

6.0-60 The Billups' panel also testified that the replacement of the Indian Point Units with fossil-fired central-station units would lead to an increase in levels of air pollution but, on Staff's

assumptions that oil-fired replacement generation would come from increased generation at existing plants at Ravenswood, Arthur Kill and Astoria and that coal-fired replacement generation would come from new plants erected at or near the present Indian Point site, would not lead to violations of air pollution standards. (Id., pp. 12-25) In addition, shutting down Indian Point could have a minor impact on vegetation as a result of increased levels of air pollution. (Id., p. 10) Constructing new coal-fired capacity could have an impact on land use and local animal and plant life in the area of the new plant. (Id., pp. 42-46)

6.0-61 The Billups' panel also testified that the taxing jurisdictions within what Staff calls the local impact area (i.e., the Village of Buchanan, Town of Cortlandt, Verplanck Fire District and Henrk Hudson School District, Id., p. 51) depend substantially upon taxes and other payments from Indian Point Units 2 and 3 for their operating revenues. (Id., p. 59) In addition, continued operation of the plants provides a significant number of jobs to persons in the local impact area and beyond. (Id., pp. 57-60) If either or both of the Units were shut down, taxing jurisdictions in the impact area would be forced to either reduce

service or increase taxes. (Id., pp. 64, 73, 75, 77) The shutdown would have an especially adverse impact on those living on fixed incomes such as the elderly. (Id., pp. 63, 73, 77)

B. ULTIMATE AND SUBSIDIARY FINDINGS OF FACT

6.0-62 With respect to the environmental consequences of a shutdown, we find that the replacement of the Indian Point Units with fossil-fired central station units would lead to an increase in levels of air pollution but, on the assumptions that oil-fired replacement generation would come from existing oil-fired plants at Ravenswood, Arthur Kill, and Astoria, and that coal-fired replacement generation would come from new plants erected at or near the present Indian Point site, we also find that the increase in the levels of air pollution would not lead to violations of air pollution standards. (Billups' Panel Testimony, pp. 12-25)

6.0-63 With respect to the regional socioeconomic consequences of a shutdown, we find that areas in the vicinity of the Indian Point Units (i.e., the Village of Buchanan, the Town of Cortlandt, the

Verplanck Fire District, and the Henrk Hudson School District) would suffer a sharp loss of taxes and the loss of a relatively significant number of jobs. (Billups' Panel Testimony, pp. 63, 64, 73, 77)

CONTENTION 6.1

6.1-1 Contention 6.1, advanced by West Branch Conservation Association (WBCA), states:

> "An economic consequence of the shutdown of Indian Point Units 2 and 3 would be an economic benefit accruing to Rockland County through the sale of replacement power."

A. INTERVENOR'S TESTIMONY

6.1-2 Appearing on behalf of WBCA, Mr. Walter L. Fleisher testified that Orange and Rockland has excess capacity, and that, if ten percent of the energy lost from a shutdown of the Indian Point Units were replaced by sales from Orange and Rockland, Rockland County would benefit by $2,077,000 per year. (Fleisher Testimony, p. 3) However, Mr. Fleisher agreed that, for the 12 months ended March 31, 1982, Orange and Rockland purchased $27 million worth of power (Tr. 14081), which exceeded the $26 million that it sold to other utilities. (Fleisher Testimony, p. 2) He also agreed that in 1981 and 1982, Orange and Rockland purchased 775,451 Mwhr and 995,493 Mwhr, respectively, (Fleisher, Tr. 14085), which is substantially more than the 510,371 Mwhr Orange and Rockland sold in the 12 months ended March 31, 1982. (Fleisher Testimony, p. 2)

B. LICENSEES' TESTIMONY

6.1-3 Mr. Meehan testified that, while Orange and Rockland would in some years receive a small benefit from a shutdown of the Indian Point plants, the overall effect would be an increase in Orange and Rockland's production costs. (Meehan Testimony, p. 29; Table 4.4, (p. 33)) That is so, Mr. Meehan explained, because Orange and Rockland is a net purchaser of economy energy and the removal of Indian Point would increase the cost of economy energy available to it. (Id.)

C. ULTIMATE AND SUBSIDIARY FINDINGS OF FACT

6.1-4 We find that the evidence mandates a rejection of Contention 6.1. Orange and Rockland is a net purchaser of economy energy, and its total production costs between the years 1984 and 1999 would increase (Meehan Testimony, p. 29; Table 4.4 (p. 33)).

CONTENTION 6.2

6.2-1 Contention 6.2 is a reformulation of a contention originally presented by Parents Concerned

About Indian Point ("Parents"). As reformulated, Contention 6.2 states:

> "A benefit would accrue from the shutdown of Indian Point Units 2 and 3 because the environment of children in the vicinity would be improved by a decrease in the release of radioactive material."

A. STAFF'S TESTIMONY

6.2-2 Appearing on behalf of Staff, Dr. Edward F. Branagan, Jr., a health physicist in the Commission's Office of Nuclear Regulation, testified that the risk to a child exposed to radioactive effluents from one reactor-year of normal operations at Indian Point is a small fraction of the risk from one year of exposure to natural background radiation; that the risk to a child is not significantly different than the risk to an average individual in a population comprised of all age groups; and that the risk to a child exposed to radioactive materials released through one reactor-year of routine operations at Indian Point is much less than the risk from exposure to any of the major sources of radiation and within the same range as the risks from exposure to many of the other common sources of enhanced radiation exposure. (Branagan Testimony, p. 11)

6.2-3 In his supplemental testimony, Dr. Branagan testified that the dose to the 50-mile population from normal operations per reactor year is 20 person-rems, that the dose to the same population from exposure to natural background radiation is 1,900,000 person-rems, and that the former is therefore but a small fraction of the latter. (Branagan Supplemental Testimony, p. 5)

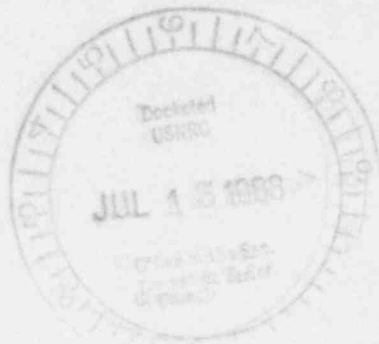
6.2-4 Using the dose of 20 person-rems and a genetic risk estimator based on models described in a report by the National Academy of Sciences, known as the BEIR I Report, Dr. Branagan estimated that about 0.005 of a potential genetic disorder may occur over all future generations of the population within 50 miles of the plant due to exposure to radioactive effluents from one reactor-year of normal operation. (Branagan Supplemental Testimony, p. 4) That, Dr. Branagan testified, is a very small part (less than 1 part in a billion) of the estimated normal incidence of genetic abnormalities. (Id, p. 6)

B. INTERVENOR'S TESTIMONY

6.2-5 In purported support of Contention 6.2, Mr. John R. Thornborough and Mr. Daryl Bohning testified that any increase in background radiation is damaging. (Thornborough and Bohning Testimony, p. 4) However, they agreed that they had not quantified the benefit to children which would be realized by closing the plants, and were not sure that the benefit could be quantified (Thornborough and Bohning, Tr. 14382); that they were aware that there are airborne releases of radioactivity from a coal-burning plant (Thornborough and Bohning, Tr. 14383-84); that the contribution of Indian Point's routine operations to background radiation "is absolutely miniscule" (Thornborough and Bohning, Tr. 14393); that the National Council for Radiation Protection and Measurement, conceded by Mr. Bohning to be a reputable organization, has recommended an occupational exposure limit of 500 mrems for pregnant women during their pregnancy (Thornborough and Bohning, Tr. 14388-14391); and that natural background radiation varies from place to place around the world by amounts significantly higher than the radiation around Indian Point (Thornborough and Bohning, Tr. 14400-14401).

C. ULTIMATE AND SUBSIDIARY FINDINGS OF FACT

6.2-6 We find that the evidence mandates a rejection of Contention 6.2. When, in our reformulation of Contention 6.2, we referred to an improvement in the environment by a decrease in the release of radioactive material, we had in mind something more than an absolutely insignificant or de minimus improvement in the environment. Accordingly, we do not have to resolve the question whether there is a threshold below which radiation has no effect. Given Staff's undisputed testimony that the risk to a child exposed to radioactive effluents from one reactor-year of normal operations at Indian Point is a small fraction of the risk from one year of exposure to natural background radiation (Branagan Testimony, p. 11), and given the testimony by Messrs. Thornborough and Bohning that the contribution of Indian Point's routine operations to background radiation "is absolutely miniscule", we could not find anything beyond an absolutely insignificant or de minimus improvement in the environment -- if we could, in fact, find that. Accordingly, Contention 6.2 must be rejected.



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) : July 11, 1983

CON EDISON'S PROPOSED OPINION, CONCLUSIONS OF LAW AND RECOMMENDATIONS

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NATURE OF PROCEEDINGS AND IDENTIFICATION OF PARTIES

This is a discretionary adjudication which was initiated by the Commission to inquire into the extent, if any, that the population density in the vicinity of Indian Point Units 2 and 3 in Buchanan, New York affects the risks posed by the plant compared to the spectrum of risks posed by other nuclear power plants licensed to operate by the Commission. In re Consolidated Edison Co. (Indian Point Units 2 and 3) 13 NRC 1 (1981)

This proceeding had its origins with a request filed by the Union of Concerned Scientists (UCS) on September 17, 1979, in which it sought an order pursuant to Section 2.206 of the Commission's Rules of Practice commencing a proceeding to modify, suspend or revoke the operating licenses for Indian Point Units 2 and 3. The petition nowhere suggested that Indian Point failed to comply in any way with applicable Commission regulations, nor did it contend that Indian Point was any less safe than other power plants located throughout the country. The UCS petition complained primarily of two things: (1) that relatively high population levels in the vicinity of Indian Point made necessary "specific added safety features and off-site emergency measures," and (2) that the Commission's "backfitting" practices should be changed to require demonstration "that specific design features in each plant provide a degree of protection equivalent to that which

would be provided by . . . new pressurized water reactors." The petition acknowledged that "some of the safety issues raised by this petition are not unique to the Indian Point nuclear plants."

The Commission treated the Petition as a request for action pursuant to 10 CFR § 2.206 and referred it to the Office of Nuclear Reactor Regulation (NRR). On February 11, 1980, the Director of NRR recommended denial of the Petition's request for a hearing and a shutdown, and he directed the licensees to undertake operational and training initiatives to further enhance plant safety. In re Consolidated Edison Co. (Indian Point, Units 1, 2, and 3), 11 N.R.C. 351 (1980).

Despite the recommendation of the Director, on February 15, 1980 the NRC published a notice which solicited public comment on the Director's decision. (See 45 Fed. Reg. 11969) Thereafter the Commission determined to conduct a special investigatory-adjudicatory proceeding before an Atomic and Safety Licensing Board, and to establish a Task Force to analyze and report to the Commission on the risk, both in terms of probabilities and consequences, of the Indian Point plant compared to the risks of other operating plants. (Order dated May 30, 1980)

On June 30, 1980, the Task Force issued its report, concluding that the overall (i.e., societal) risk posed by Indian Point was about the same as the typical reactor at a typical site. Additionally, the Task Force found that while the Indian Point site was more risky than the average site due to the high population density in the vicinity of Indian Point, the design features of these two units reduce the accident risk of Indian Point by a comparable factor. With respect to the risk posed to individuals living in the vicinity of Indian Point, the Task Force concluded that the Indian Point units posed 30 to 50 times less risk to persons and property than the postulated typical (Surry) reactor.

On January 8, 1981, the Commission ordered that special proceedings be conducted by an Atomic Safety and Licensing Board (Board) and permitted continued operation of the plants during the hearings. In re Consolidated Edison Co. (Indian Point, Units 2 and 3), 13 NRC 1 (1981). Although the hearing was to use "the full procedural format of a trial-type adjudication, including discovery and cross-examination," the Commission stated that the proceeding was not to be an "on the record" proceeding, and exempted the hearing from the application of certain of the important procedural requirements of 10 CFR Part 2 (1981). In re Consolidated Edison Co., 13 NRC at 5 & n.4. The Commission emphasized that its "primary concern is the

extent to which the population around Indian Point affects the risk posed by Indian Point as compared to the spectrum of risks posed by other nuclear plants. Id. at 6. The Commission also asked that the Board inquire into non-risk issues of shutdown impacts and emergency planning. The Commission indicated that the Board should not attempt to make a final decision, but should forward the record of the proceeding, with recommendations, to the Commission for final agency action.

On September 18,1981, the NRC revised the questions it had posed in its January 8 order to more explicitly require equal attention to probabilities and consequences and empaneled a Licensing Board. In re Consolidated Edison Co. (Indian Point Units 2 and 3), 14 NRC 610 (1981). On October 7, 1981 the Board ordered that requests to be admitted as parties and to intervene be filed by November 6, 1981. (See Order dated October 1, 1981) After various prehearing conferences the Board began hearing witnesses on June 22, 1982. On July 27, 1982, the Board was directed by the Commission to reconsider the 22 contentions it had admitted into the proceeding. Subsequently, on August 9, 1982 the Board sought further Commission guidance on formulation of contentions. On October 1, 1982, the Board issued reformulated contentions. Following a prehearing conference on November 3 and 4, 1982, the Board issued its final determination on

contentions on non-emergency planning issues and established a schedule for discovery, the filing of testimony and hearing. (See Order dated November 15, 1982) Hearings recommenced on January 10, 1983 and were completed on April 29, 1983.

Parties

In addition to the Commission Staff, the parties to this proceeding are Consolidated Edison Edison Company of New York, Inc. (Con Edison), and the Power Authority of the State of New York (Power Authority), licensees of Indian Point Units 2 and 3, respectively (licensees). The following participated as intervenors: Richard L. Brodsky (Brodsky), Friends of the Earth (FOE), the Greater New York Council on Energy (GNYCE), the New York City Audubon Society (Audubon), Parents Concerned About Indian Point (Parents), Rockland Cirizens for Safe Energy (RSCE), the Union of Concerned Scientists, Inc./New York Public Interest Research Group, Inc. (UCS/NYPIRG), the West Branch Conservation Association (WBCA), and the Westchester People's Action Coalition (WESPAC). The following participated as interested states: the Attorney General of the State of New York (Attorney General), the New York State Energy Office (Energy Office), the Westchester County Executive Alfred Del Bello (Del Bello), the Metropolitan

Transportation Authority (MTA), members of the Council of the City of New York (NYC Council Members), the Port Authority of the State of New York and New Jersey (Port Authority), the County of Rockland (Rockland), the New York State Assembly and its Special Committee on Nuclear Power Safety (State Assembly), and the Village of Buchanan (Buchanan).

STATEMENT OF ISSUES (COMMISSION QUESTIONS)

As finally formulated by the Commission in its Order of September 18, 1981, the Questions to be addressed are as follows:

Commission Question 1

> What risk may be posed by serious accidents at Indian Point 2 and 3, including accidents not considered in the plants' design basis, pending and after any improvements described in (2) and(4) below? Although not requiring the preparation of an Environmental Impact Statement, the Commission intends that the review with respect to this question be conducted consistent with the guidance provided the staff in the Statement of Interim Policy on "Nuclear Power Plant Accident Considerations under the National Environmental Policy Act of 1969;" 44 FR 40101 (June 13,1980). (footnote omitted)

Commission Question 2

> What improvement in the level of safety will result from measures required or referenced in the Director's Order to the licencee, dated February 11, 1980? (A contention by a party that one or more specific safety measures, in addition to those identified or referenced by the Director, should be required as a condition of operation would be within the scope of this inquiry if, according to the Licensing Board, admission of the contention seems likely to be important to resolving whether (a) there exists a significant risk to public health and safety, notwithstanding the Director's measures, and (b) the additional proposed measures would result in a significant reduction in that risk.)

Commission Question 3

> What is the current status and degree of conformance with NRC/FEMA guidelines of

state and local emergency planning within a 10-mile radius of the site and, of the extent that it is relevant to risks posed by the two plants, beyond a 10-mile radius? In this context, an effort should be made to establish what the minimum number of hours warning for an effective evacuation a 10-mile quadrant at Indian Point would be. The FEMA position should be taken as a rebuttable presumption for this estimate.

Commission Question 4

What improvements in the level of emergency planning can be expected in the near future, and on what time schedule, and are other specific offsite emergency procedures that are feasible and should be taken to protect the public?

Commission Question 5

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.)

Commission Question 6

What would be the energy, environmental, economic or other consequences of a shut-down of Indian Point Unit 2 and/or Unit 3?

OPINION ON INDIVIDUAL ISSUES

The Board discusses and resolves each of the Commission's Questions, the Contentions, and the Board Questions below. Because of the lengthy number of witnesses who appeared, witnesses have been identified in Attachment A to this Opinion, grouped by the Board Question to which their testimony pertains.

THE RISK OF SERIOUS ACCIDENTS AT INDIAN POINT

Commission Question 1 asked the Board to address the risk of serious accidents, including beyond design basis accidents, at Indian Point Units 2 and 3. Under the Commission's January 8, 1981 and September 18, 1981 Orders, this inquiry was to employ quantitative risk assessment calculations, and to be conducted consistent with the Commission's June 1980 Interim Policy on nuclear power plant accident considerations. One contention was admitted as an issue under this Commission question, and the various parties were also invited to submit testimony on four related Board questions. Because Commission Question 1, Contention 1.1, and Board Question 1.1 raise intertwined issues, we shall discuss them jointly.

The licensees and the Staff offered testimony derived from the Indian Point Probabilistic Safety Study (IPPSS), a probabilistic risk assessment of the Indian Point units which was commissioned by the licensees and first published in March 1982. The sponsors of Contention 1.1 alleged that health and property risks would be created at considerable distances beyond the approximately ten mile plume exposure pathway Emergency Planning Zone (EPZ) by a serious accident at the plant. Our Board questions asked the parties specifically to address the probability of occurrence as well as the consequences of serious accidents

at the Indian Point units, a letter report evaluating the IPPSS prepared by the Sandia National Laboratory, the bearing upon Indian Point risk of a 1982 NRC study entitled "Precursors to Potential Severe Core Damage Accidents: 1969-79, A Status Report," the probabilities associated with an Indian Point accident consequence study commissioned by certain intervenors and interested states, and the risk associated with pressurized thermal shock to the reactor pressure vessels.

Licensees presented testimony that the likelihood of accidents at the Indian Point units which would pose risks to the health and safety of the surrounding population is exceedingly remote. Testimony on this issue from intervenors and interested states questioned the techniques used in developing quantitative assessments of risk, and alleged that the consequences of a serious accident at Indian Point if one were to occur could have a substantial impact upon New York City and other areas at considerable distances form the plant.

Probabilistic Risk Assessment

Quantitative assessment of the risk of serious accidents at nuclear power plants has to date not been a part of the Commission's licensing process. To our knowledge, this is the first occasion in which a Licensing

Board has been asked to evaluate numerical estimates of the risk posed by a nuclear plant. Risk, as expressed quantitatively in this proceeding, consists of two distinct but related components: the probability, or likelihood, of the occurrence of a particular event, and the severity of the outcome of that event. Both of these elements must be considered in conjunction with one another before there can be an accurate and complete portrayal of risk. (Finding 1.1-6)

In nuclear plant risk assessment, there are several standard expressions of risk normally used. The first is individual risk, the likelihood that a particular person in the vicinity of a plant would be affected by a serious accident at that facility. The second is societal risk, which expresses the likelihood that a population of people would be affected by an accident. (Finding 1.1-7) Both individual and societal risks are expressed in terms of distinct measures of impact such as early fatalities, radiation illnesses, thyroid cancers, latent cancer fatalities, and total population dose. (Findings 1.1-7 and 1.1-216) Property damage, while not strictly a risk to public health, is also used as a measure of plant risk.

Although new to Atomic Safety and Licensing Board proceedings, probabilistic risk assessment (PRA) techniques are not new to the Nuclear Regulatory Commission. The first comprehensive nuclear plant risk assessment, known

variously as the Reactor Safety Study, Rasmussen Report, and WASH-1400, was published by the Commission in 1975. The process itself has developed rapidly in recent years, and has been applied to more than two dozen nuclear plants, as well as the space program, chemical plants, commercial aircraft, and other technologically complex systems. (Finding 1.1-11 and 12) PRA is an advanced technique for quantifying risk, using mathematical models which result in explicit numerical calculations of the probabilities and consequences, i.e. risks, of accidental events. (Finding 1.1-9) PRA is the only method of nuclear plant safety analysis that accounts in a comprehensive and integrated way for the cascading effects of multiple initiating events and component failures, interactions among systems, and human error. (Id.) PRA also provides a framework which can accommodate all aspects of reactor accident susceptibility, and while it has both strengths and weaknesses, there was general agreement among the parties offering risk testimony that PRA provides the best estimates of risk which can be derived from modeling nuclear plant systems. (Id.)

Whenever efforts are made to attach numerical likelihoods to future events, there will be uncertainty as to the accuracy and correctness of the results achieved by the analysis. As described below, the IPPSS study employed novel techniques for portraying the uncertainties

associated with the risk estimates, and numerically expressing a degree of confidence in frequency values. (Finding 1.1-18) A number of intervenors contended that too great of an uncertainty as to the likelihood of a major accident is inherent in the techniques of PRA, and that accordingly little confidence can be placed in the accuracy of the risk estimates which are obtained. Nonetheless, the decision to use probabilistic tools is not one that the Board is free to reject. In directing that this proceeding be conducted the Commission stated that:

> "In directing a comparision of the risks of the Indian Point units with those from a representative group of other operating units, the Commission is fully aware of the uncertainties that attend such quantitative risk assessment calculations (reference NUREG-CR-0400, the Lewis Report, and the Commission policy statement on it.) Despite these uncertainties, risk assessment methods offer the best means available for objective and quantitative comparison of the kind needed here." 13 NRC 1, 6, CLI-81-1 (1981)

The Board has accordingly been mindful of the need to incorporate considerations of uncertainty into the risk analyses which have been presented. We are of the view that uncertainty in the risk estimates has been adequately addressed.

THE INDIAN POINT PROBABILISTIC SAFETY STUDY IPPSS)

Prior to its publication in March 1982, work on the Indian Point Probabilistic Safety Study required approximately two years and 50 man years of effort. (Finding 1.1-17) An amendment to IPPSS was subsequently submitted to the NRC in January 1983, and its results were included in licensees' testimony. (Id.) The study was performed by numerous consultants, working in concert with the licensees to see that appropriate plant-specific and site-specific data were included. (Id.)

METHODOLOGY

The first step in preparing a probabilistic assessment is the identification of initiating events, i.e., disturbances in reactor operation that have the potential to lead to releases of radiation were it not for the intervention or presence of safety systems. Such events require a rapid reactor shutdown and subsequent safety system operation to prevent core melt. "Internal" initiating events are those which originate with malfunctions or failures of plant equipment or systems, including those caused by human error. "External" initiating events are those that originate from other

causes, including earthquakes, fires, floods and high winds. (Finding 1.1-24)

"Event trees" and "fault trees" are used in the quantification of initiating events, plant response and containment response. An event tree is a diagram that depicts the possible sequences of system functions and operator actions that could result from a given initiating event, including those which could possibly lead to core damage or to containment failure. The branches in the event trees represent the success or failure of the various safety systems or operator action in the plant. Fault trees are models of a system's reliability. They trace the origin of system failures to component failures and human errors. (Finding 1.1-25)

The catalogs of initiating events, event trees and fault trees are assembled into a mathematical plant response model that permits accident sequence likelihoods to be calculated from initiating event frequencies, component failure rates and human action and error rates. This plant response model delineates the system functions needed to prevent core damage and to protect the containment. From the plant response analysis, groups of accident sequences with similar effects on the plant are combined into a reduced number of "plant damage states," and their total frequencies are determined. These physical conditions which are the results of the plants' response

analysis are then employed in a containment response analysis. The end point of these calculations is an evaluation of whether or not the containment building fails, how it fails, when it fails and what quantity of radioactivity and energy escaped from it to the environment. (Findings 1.1-26 and 27)

The consequence analysis is the assessment of public health and economic impacts from postulated releases of radioactive material. The consequence analysis estimates the probability of various health effects from exposure to the releases assigned to each release category, given the occurrence of the release. The results of the consequence analysis are used along with the release frequencies determined in the plant and containment response analyses, to construct final risk curves. (Finding 1.1-28)

In developing its estimates of risk, the Staff employed point estimates which were used to arrive at single-risk curves called Complementary Cumulative Distribution Functions (CCDF). The authors of the IPPSS employed families of curves rather than point estimates to account for uncertainties associated with the risk estimates, and to better represent the degree of confidence in the frequency numbers. Although the Staff did not use the family of curves approach, the Staff testified that curves are more realistic than point estimates because they account for uncertainties. (Findings 1.1-29 and 30)

The IPPSS employed a Bayesian method to update and integrate nuclear plant data. The use of Bayesian methodology is appropriate in such an analysis because it is a mathematical formula which allows quantification and integration of three important types of information: (1) general engineering knowledge and judgment; (2) industry-wide data; and (3) plant-specific data. (Finding 1.1-31) The chief controversy over the use of Bayesian methodology concerns the use and choices of prior distributions. A prior distribution expresses the state of knowledge prior to observing data. A Staff witness recommended a different analytical method, the capitalized maximus method, which is non-Bayesian and relies only on plant-specific data. However, when Sandia National Laboratories calculated important IPPSS sequences using the maximus method, the differences in results were characterized by a Staff witness as not being statistically significant. (Findings 1.1-32 and 33) Another Staff witness testified that in view of the small differences in results between alternate methods, the controversy over a proper statistical method was not important to the risk calculations. (Finding 1.1-33) The Board agrees, based on the record before it, that the Bayesian method was appropriate for the purposes for which it was used in IPPSS. The mathematical approach used in IPPSS was subjected to critical review by Sandia, which found no appreciable effect or results. (Finding 1.1-34)

An intervenor witness did not object to the use of Bayesian methodology, but did object to the frequent use of a lognormal distribution as a prior. However, lognormal distributions are used extensively in risk work and closely conform to the underlying distribution of frequencies. (Findings 1.1-35 and 1.1-37) This witness also questioned the adoption of a definite prior distribution and recommended a "k" factor, which adjusts the data by accounting for various levels of indeterminacy. However, the use of such a factor did not have a significant effect upon failure rates. (Finding 1.1-36) The witness also proposed the use of a "minimax" theory, which ignores probabilities and contrasts the worst possible consequences of operating a nuclear plant such as Indian Point with the worst possible consequences of alternatives. The "minimax" theory as proposed by the witness has never been employed in public decision making, and in any event the Board must reject it because we have been asked by the Commission to use probabilistic techniques giving equal weight to probabilities and consequences alike. (Findings 1.1-38 and 39)

Another methodological requirement by any PRA is the treatment of uncertainty in the results achieved. The IPPSS authors pioneered a technique for the propagation of uncertainties which can accommodate the effects of modeling approximations, phenomenological uncertainties and completeness, as well as the more commonly treated statis-

tical uncertainties. (Findings 1.1-18 and 1.1-42) The IPPSS is the most comprehensive quantitative assessment of uncertainties that has been presented in a PRA to date. Where there is no generally accepted rigorous mathematical model for uncertainty analysis, the approach used by the IPPSS authors to quantify uncertainty is on the forefront of the technology. (Id.) Staff testified that the treatment of uncertainty in IPPSS was a reasonable way to proceed. Staff's attempt to use a different technique for quantifying uncertainty than that used in IPPSS was unsuccessful. (Finding 1.1-43) The uncertainties in the IPPSS appear large because uncertainties are proportionately greater for very small probabilities than they are for very large probabilities. (1.1-44)

The Board finds that the use of Bayesian methodology in IPPSS was reasonable and appropriate method to use which did not produce appreciably different results when compared to alternate methodologies. The weight of the record supports the conclusion that it has advantages over other methodologies due to its ability to embrace and assimilate large amounts of useful data. The treatment of uncertainty in IPPSS was also reasonable and appropriate, and it is uncontroverted that it is at the forefront of efforts to represent mathematical uncertainty in a comprehensive manner.

INITIATING EVENTS

Initiating events, such as operator error and earthquakes, are occurrences either internal or external to the nuclear plant system which may cause a departure from normal plant operation. Some initiating events have the potential to cause severe core damage unless rapid reactor shutdown and subsequent safety system operation occur.

Internal initiating events include loss of feedwater flow, loss of AC or DC power, turbine trip, operator error, and pipe failures. (Finding 1.1-75) In IPPSS, internal initiating events were identified through the use of a number of sources, including the Final Safety Analysis Report (FSAR), Licensee Event Reports (LERs), EPRI reports, the RSS, various Westinghouse studies, and Indian Point plant-specific reports. (Finding 1.1-76) Staff concluded that the internal initiating events included in IPPSS are relatively complete and their frequency estimates are reasonable compared to those addressed in previous PRAs, and an intervenor witness also acknowledged that IPPSS initiating event frequencies are reasonable and quite conservatively high. (Findings 1.1-77 and 78)

Among the external initiating events considered in IPPSS were earthquakes, fires, turbine missiles, flooding from plant system and external sources, wind and wind-induced missiles, and transportation and storage of

hazardous materials. The most significant external initiating events are earthquakes, wind and fires. External initiating events were compiled by the authors of IPPSS using the FSARs as a basis and updating them with information and data bases from federal and local government agencies, and from appropriate industrial and commerical organizations. (Finding 1.1-79)

1. Earthquakes

It was conceded by all witnesses who addressed the topic that the IPPSS earthquake risk analysis has broken new ground in its methodology and use of advanced data. The IPPSS methodology is discussed in the PRA Procedures Guide, NUREG/CR-2300. (Finding 1.1-80) In the IPPSS as initially published, the core melt frequency for Indian Point Unit 2 was dominated by seismically induced interaction between the Unit 1 superheater building and the Unit 2 control building. Since the IPPSS was initially published, after investigating various ways of reducing the core-melt contribution from this event, Con Edison decided to change the physical configuration by widening the clearance between the two buildings at the roof line, and by inserting rubber pads in the gap between the two buildings to absorb impact energy in the unlikely event of an earthquake that would cause the accident scenario revealed in the IPPSS. The authors of IPPSS have

re-examined the seismic contribution to overall mean core melt frequency for Unit 2 taking into account the modification, and that frequency has been reduced from 1.4 x 10^{-4} to 7.9 x 10^{-6}. (Finding 1.1-81)

A recent reanalysis of the ability of Indian Point containments to withstand severe earthquakes (seismic fragility), and a comparison with the largest earthquake that can be reasonably postulated for this area, establishes that direct failure of the Indian Point containment buildings as a result of a seismic event is not a credible event. The Indian Point containments have a median structural capacity of 2.7 g. Use of ground acceleration (g) is the appropriate method of assessing the seismic capacity of a structure. This involves determining the earthquake's magnitude or intensity and then calculating the attenuation from the potential source point to the plant and the acceleration at the plant. The maximum acceleration associated with the largest earthquake that can be reasonably postulated for this area is considerably below the structural capability of the containment. Accordingly, the IPPSS release category caused by a massive seismic event and described in IPPSS is no longer applicable. (Findings 1.1-82 and 83) Although this analysis shows that the containment cannot directly fail from any seismic event which could occur in the area, Staff has retained a failure mode for this accident until a

detailed review of the recent analysis is completed, which had not occurred by the time the hearings were concluded. The overall risk is reduced with elimination of this failure mode, since half of the early fatality risk for Indian Point Unit 2 had come from that release category. (Finding 1.1-84)

The largest acceleration expected in the vicinity of Indian Point within a 500-year period is less than .1g. Earthquakes with ground accelerations of about .1g to .3g could result in injuries and loss of life to the public from heavy damage to conventional structures such as residences and office buildings. However, such accelerations would not be great enough to threaten even the weakest crtitical structure or equipment at the Indian Point plants. (Finding 1.1-85)

The safety related structures and equipment of Indian Point Units 2 and 3 are designed to withstand without loss of function the design basis earthquake of 0.15g. Detailed studies of these safety related structures and equipment show that their actual capability to withstand seismic events considerably exceeds the design basis value. The lowest median acceleration capacities for safety related equipment are in the range of 0.7g to 0.9g. Even the least capable of these components is likely to function during very large earthquakes. Safety related equipment with even higher median acceleration capacities

would still be available to limit the consequences of seismically induced failures. (Finding 1.1-89) Sandia National Laboratories' initial concern regarding the seismic capacity of the control room ceiling was explicitly addressed in Amendment 1 to IPPSS. Reanalysis and modifications to the ceilings resulted in the conclusion that they no longer contribute significantly to seismic risk. (Finding 1.1-90)

Testimony at the hearings indicated that significant conservatisms (i.e overestatements of risk) existed in the IPPSS seismic analysis. These conservatisms exist because of the distributions of data used and because IPPSS made "pessimistic" rather than "realistic" assumptions in the seismic analysis, particularly regarding the structural response to ground motion. Notwithstanding these overstatements, independent analyses of Staff's consultant regarding the estimates of seismic risk at Indian Point confirm that no significant differences exist between the results of such analyses and those in IPPSS. (Findings 1.1-91 through 1.1-93)

2. Fires

The IPPSS fire analysis began with a preliminary analysis of potential fires throughout the plant, followed by a more detailed, on-site evaluation for those critical locations where a fire might damage important equipment or

cables, and thus initiate an accident scenario. The authors of IPPSS believed that very detailed analyses of all plant locations would not significantly change the results. (Finding 1.1-95) An intervenor disagreed and argued that more locations should have been analyzed. (Finding 1.1-96) When the fire risk was reanalyzed to account for plant modifications made subsequent to the initial publication of IPPSS, locations that had not been originally evaluated in detail were explicitly quantified. No significant changes resulted from this reanalysis. (Finding 1.1-96)

According to Staff, control room fires should have been comprehensively analyzed in IPPSS because of the potentially large consequences which could result from such events. However, Staff stated that there are no data to support a high estimated frequency of serious control room fires. Licensees provided a more detailed analysis of potential fires in this area of the plant in Amendment 1 to IPPSS. Because this area is continuously manned, a fire would be detected almost immediately and attempts to extinguish it would begin promptly. The results of this analysis showed that the freqency of core melt accidents originating from control room fires accounts for only a small fraction of the other fire-initiated core melts. (Finding 1.1-97) Fires in critical locations are unlikely at Indian Point because of such precautions as the emphasis

on nonflammable materials, administrative controls, fire barriers, and fire detection and suppression systems. Fires involving large quantities of combustible materials in critical locations are the only ones that could result in serious damage to the plant, however there are no such conditions at the plant. Finding 1.1-98)

Appendix R to 10 C.F.R. Part 50, which became effective on February 17, 1981, provides new licensing requirements to reduce fire risk. Implementation and enforcement of Appendix R is in progress for all nuclear power plants, including Indian Point Units 2 and 3. Modifications at the Indian Point units include the capability to supply power to various safety components by routing additional cables which are physically separated from normal cable routes. Because of this additional cable routing, fires in locations such as the cable spreading room, the switchgear room, the electrical tunnel and penetration area, and the motor control center area are far less likely to have any significant risk implications. The record reflects that these changes were initiated at Unit 2 by the licensee, and that they yield a reduction in mean core melt frequency for five fire sequences of approximately one order of magnitude at Unit 2. (Findings 1.1-03 and 104)

It is not disputed in the record that the IPPSS evaluation of fires was performed in a competent manner and

appears to have identified the critical plant areas where a fire can cause an initiating event and simultaneously fail redundant safety systems which may be required to function in response to the initiating event. Differences between Staff and IPPSS estimates of mean core melt frequencies for fire-initiated sequences are small. (Finding 1.1-107)

3. Wind

The most vulnerable critical structures at Indian Point have wind capacities of well over 100 mph, most over 200 mph. Forces associated with extreme wind sufficiently intense to cause damage to critical structures at Indian Point would be very rare. Winds exceeding 100 mph at the site have mean occurrence intervals of 10,000 years. Winds exceeding 150 mph have mean occurrence intervals of over 100,000 years. Such winds would cause widespread offsite damage as a direct result of the event before there would be any failures at Indian Point. Extreme winds are not a significant contribution to risk at Indian Point Unit 3. While hurricanes were originally thought to be a significant contributor to risk at Indian Point Unit 2, a technical specification has now been implemented which requires the facility to shut down in advance of an approaching severe hurricane. Offsite radiological risk can be largely eliminated by shutting down Indian Point Unit 2 some hours ahead of the arrival of a severe

hurricane. Accordingly, with the new technical specification in place, hurricanes no longer constitute a significant contribution to risk at Indian Point 2. (Findings 1.1-109 and 110) Regarding tornados, the Staff reviewed the history of tornados in New York State. It concluded that the IPPSS numbers constitute a fairly representative, if not conservative analysis of the tornado threat at Indian Point. The analysis shows that tornados, while not negligible contributors to risk at Indian Point, are not a significant contributor. (Finding 1.1-111)

4. Other external initiating events

The overall likelihood of floods from external sources capable of threatening the safety of the Indian Point plants would be once per 10,000 reactor years or less. Even if water levels were high enough to breach the plant's boundaries, core melt would not necessarily occur because the plant should be able to withstand this condition. Moreover, extra protection would be provided because the plant would have been in a shutdown condition for some time if high water were threatening. The frequency of water levels exceeding the grade elevation of the buildings at Indian Point is somewhere between once in a hundred million years and once in a trillion years. Floods are therefore not significant contributors to initiating core melt accidents at Indian Point. (Finding

1.1-112) The overall contribution to risk from airplane crashes, pipeline failures and barge accidents are quite small compared to other external events. (Findings 1.1-113 and 114)

PLANT RESPONSE

IPPSS plant analysis is particularly sophisticated and refined. Because plant response to internal and external initiating events is similar, the same model was used to analyze both. The basic plant response model delineates the system functions necessary to avoid core damage and to protect the containment, and was used to calculate the probability that these functions are successfully provided. For the data base used to model plant response, the authors of IPPSS collected industry-wide data and used it to provide initial probability distributions or generic distributions. These distributions were then supplemented with specific data from Indian Point. Four types of data were used: component failure data, maintenance and testing data, human error rates, and initiating event frequency data. Bayes' theorem was used to integrate the industry-wide data with the plant-specific data. (Finding 1.115 through 1.1-117)

The IPPSS evaluated plant response to internal initiating events in terms of an overall event tree model

supported by a detailed analysis to determine the reliability of the systems that must operate to accomplish the safety function identified in the plant event trees. The IPPSS considered system dependencies, including dependencies on auxiliary, shared, or connected systems. This analysis included consideration of states of electric power, and of the availability of auxiliary systems, interfacing systems, and actuation signals. System fault trees using plant data were employed to identify and display combinations of components whose failure could cause system failure. (Finding 1.1-121)

The plant-specific and industry-wide data bases upon which IPPSS relied for failure rates include information regarding equipment and components of varying ages. Failure rates derived from this data base thus include "burn-in" and "wear-out" failures. Therefore, using a constant, or average, failure rate with appropriate uncertainties adequately accounts for aging effects. (Finding 1.1-50)

The results of the plant response analysis are determinations of frequency of core melt. Core melt frequency is a poor indicator of public health risk because there is no correlation between the magnitude of the core melt frequency and the magnitude of the risk. For example, 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 radioactivity. Additionally, approximately 95 percent of the calculated early fatality risk is due to the interfacing systems LOCA, which contributes less than one-half of one percent to the core melt frequency. (Finding 1.1-127)

Licensees estimated the median frequency of core melt at Indian Point caused by internally initiated events to be once in 20,000 reactor years and once in 7,100 reactor years for all initiating events (internal and external). The comparable values for Indian Point 3 are once in 33,000 and once in 20,000 reactor years. The Staff's best estimate of the core melt frequency is once in 3,000 reactor years for both plants when both internal and external initiating events are considered. However, the Staff acknowledged that their estimate was conservative (estimated a more frequent occurrence) since they did not consider the strengthening of the control room ceiling panels nor did they eliminate the conservatisms in the frequency of hurricane-induced severe reactor accidents. (Finding 1.1-128)

CONTAINMENT REPONSE

Containment response analysis involves investigating the phenomena of reactor core melt accidents, and assessing the response of the containment to these phenomena. The authors of IPPSS evaluated degraded core

phenomenology and containment capability, and developed detailed containment event trees. IPPSS containment response analysis, which includes a more comprehensive investigation of core melt phenomenology than that in the RSS, demonstrated that in general containment failure probabilities for the Indian Point units are much lower than had previously been assumed. The major conclusions of the licensees' containment analyses were substantiated by Staff's independent analyses. (Finding 1.1-130)

The plant damage states derived from the plant analysis are the starting points for the containment event tree. Branch points on the containment event tree represent success or failure probabilities based on physical laws, as well as on related experiments, analyses and engineering judgment. Detailed evaluations were performed by the authors of IPPSS and by Staff for those branch points which are most important to containment response. The end point of the containment event tree is a set of radionuclide release categories which are used in the consequence analysis. Over 6,000 accident progressions were considered in the containment event tree in connection with the preparation of IPPSS. (Finding 1.1-131 and 132)

The principal issue in the hearings in the area of containment analysis was the assessment of containment strength, i.e. the ultimate failure pressure. We consider that issue below in connection with the critical assessment of IPPSS.

Other significant factors in the containment analysis are steam explosions and missiles, hydrogen generation, and basement failure. Steam explosions and missiles were considered important contributions to early containment failure until recent research showed that reactor vessel geometry precluded it. Although steam explosions are now considered highly unlikely, they are conservatively accounted for in IPPSS and the Staff's analysis. (Finding 1.159 and 160) Conditions for hydrogen detonation could not be reached and were consequently not modeled in IPPSS. (Finding 1.1-161) Hydrogen burns would not reach containment failure pressure and are thus not contributors to releases at Indian Point. (Finding 1.1-168)

Staff and licensees disagreed about the amount of hydrogen that could be generated at various stages of an accident. Licensees cited analyses demonstrating that the manner in which the core moves to the lower reactor vessel and the fact that particles formed during the melt processs would be sufficiently coarse to rapidly cool the debris, which would minimize hydrogen generation. Staff testified that there is uncertainty regarding the manner in which the core would move and that particle size could be smaller than reported in the IPPSS. Citing their "lack of knowledge in this area," Staff assumed that 100 percent of the zirconium would be oxidized during in-vessel core

heatup and meltdown. They further assumed that 100 percent of the core material would exit the vessel during release to the reactor cavity, thereby producing more hydrogen. Licensees' experts believes this latter phenomenon is mechanistically impossible. (Findings 1.1-163 through 167) Staff's assumptions appear to be unrealistically conservative.

Under conditions of vessel failure without water present in the reactor cavity, the concrete basemat beneath the reactor vessel would be subject to ablation by hot molten core debris coming into contact with it. Staff believes that even with a dry cavity, complete basemat penetration will not always occur. For the dry-cavity case, IPPSS assumed that the containment would always fail by overpressurization prior to basemat penetration. Because the health consequences of an airborne release exceed those of a basemat failure, such an assumption leads to an overstatement of the risk. (Finding 1.1-174)

At the Indian Point plant, the basic geometry of the containment floor provides for the presence of water in the reactor cavity prior to a breach of the vessel, the continued availability of cooling water when containment safeguards are operational, and sufficient surface area for cooling. If water is present on a continual basis, a coolable debris bed would form and no substantial concrete decomposition would be expected to occur. (Findings 1.1-175 and 176)

The Board finds that containment failure through steam or hydrogen pressure surges, internally generated missiles, or basemat penentration, are very unlikely and are not significant contributors to risk. In addition, IPPSS presents a more realistic analysis of hydrogen behavior.

ACCIDENT CONSEQUENCE ANALYSIS

Accident consequence analysis is the assessment of public health and economic impacts from postulated releases of radioactive material. Those scenarios which lead to a release from containment to the atmosphere or which bypass containment were analyzed in great detail in both the licensees' and Staff's risk analyses. Meaningful consequence analysis of the Indian Point plants and site must be based on plant and site-specific data such as the characteristics of plant releases, meteorology, demography and topography, and protective measures such as evacuation and sheltering. (Findings 1.1-188 and 189) Licensees and Staff evaluated accident consequences for: (a) early fatalities; (b) early injuries (radiation illnesses); (c) delayed cancer fatalities, fatal and non-fatal; (d) delayed thyroid cancer cases, fatal and non-fatal; (e) population dose (man-rem) to the exposed populations; and (f) offsite economic costs. (Finding 1.1-190)

The key parameters that characterize the various hypothetical accident sequences are: (a) the magnitude of radioactive releases; (b) time of release; (c) duration of release; (d) warning time, that is, the time from the operator's determination that a release will occur until the time the release does occur; (e) height of release; and (f) the energy content of the release. Meaningful consequence analysis of the Indian Point site must be based on site specific data such as meteorology, demography, and topography, and on protective measures such as evacuation and sheltering. (Findings 1.1-189, 1.1-192 and 1.1-205)

1. Analytical Procedures and Dispersion Analyses

Both the licensees and the Staff used complex computer codes in their consequence assessment analyses. These codes are modified versions of the Calculation of Reactor Accident Consequences (CRAC) computer program used in the Reactor Safety Study (RSS). (Finding 1.1-194)

Because the RSS was applied to a composite site, the authors of IPPSS developed an improved version of the CRAC program, called Calculation of Reactor Accident Consequences Including Trajectories (CRACIT), which allowed them to fully reflect conditions specific to Indian Point. (Finding 1.1-195) Staff modified the CRAC code in certain area to allow for site-specific and problem-specific

applications.* (Finding 1.1-195) The CRACIT code, which has been used in a number of site-specific risk studies, is more complex and realistic than is the CRAC computer code. (Finding 1.1-198)

The analysis of the health effects of radioactive materials released depends upon their location, which may be either airborne or on the ground; their concentration; and the location and travel paths during evacuation of the exposed population. The consequences of a postulated accident can vary with the timing and the duration of the radioactive release. The most important factors affecting this variation relate to weater conditions at the time of the release and during dispersion. Wind speed and direction, atmospheric mixing, and precipitation govern the area affected and the concentrations of radioactive

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* The four principal differences between the CRACIT and CRAC codes are: (a) CRACIT models changes in plume trajectory, whereas CRAC models straight-line plume trajectories; (b) CRACIT models evacuation at a variety of speeds based on an analysis of the roadway network, whereas CRAC models evacuation radially away from the reactor site at a constant speed; (c) CRACIT is capable of modeling releases in up to four separate phases, whereas CRAC models all releases as a single "puff"; and (d) CRACIT meteorological sampling consists of 288 individual random samples whereas CRAC combined 91 individual random samples with 16 direction sector samples, resulting in a total of 1456 scenarios for the calculation of probability distributions for consequences. (Finding 1.1-196)

material at any given location. The modeling of protective actions can have an important influence on consequence estimates, particularly with respect to early illnesses and fatalities. (Finding 1.1-192 and 193)

In licensees' and Staff's consequence analyses, the distribution of airborne and deposited radioactive material in the environment was calculated as a function of time. The movement and shielding of people were taken into account to calculate the resulting dose and number of health effects. Many weather conditions were studied for any single postulated release to determine the probability distribution for each health effect analyzed. Assumed accident start times were selected such that they were uniformly distributed over all months and over day and night conditions. Meteorological data were selected from the data base for corresponding times. Additionally, in the IPPSS, low probability weather conditions, which would result in the most severe consequences, were specifically included through examination of the entire meteorological data base. (Finding 1.1-199)

Dispersion analysis is also an important component of both the licensees' and Staff's computer codes. The following phenomena were modeled by both parties: (a) atmospheric dispersion of released materials; (b) wet and dry deposition; (c) plume rise; and (d) inversion layers. Licensees' analysis also included detailed terrain modeling of the Indian Point vicinity. (Finding 1.1-202)

Because of the manner in which containment will fail during an overpressurization accident, radiological releases occur over an extended period of time up to several hours. Staff agreed that it is more realistic to model these long duration releases as four distinct "puffs," as was done in IPPSS. (Findings 1.1-207 and 208)

2. Plant Site Data Used in Analyses

Both licensees and Staff used plant-specific data in their respective analyses. In IPPSS and in Staff's analysis, weather data from a representative one-year period were used. Wind speed, wind direction, atmospheric stability, precipitation, and inversion layer height were the parameters assessed. IPPSS data were drawn from multiple locations near the site, and, thus, any plume that is projected to leave the site area is assumed to be influenced by weather data measured in an adjacent region. (Findings 1.1-232 and 233)

The testimony presented by Con Edison witness Lester A. Cohen strongly suggests that, because of the meteorology of the area around Indian Point, which is heavily influenced by local topography, the possibility of a plume being carried south of the EPZ is remote. This is primarily due to the light transport winds, the diurnal flow in the valley, and the topographic features around Indian Point. Because intervenors' witnesses on this

subject have neither training nor experience in meteorology, their testimony has little probative value. (Findings 1.1-233 and 4.1-10)

3. Dose Assessment Methodology and Health Effects

In addressing the methodology of dose assessment, IPPSS slightly modified the dose assessment methodology used in the RSS, to be approximately equivalent to mid-range estimates developed by the 1980 BEIR Committee. (Finding 1.1-217)

Exposure can be either early or chronic. Chronic exposure involves lower dose rates and longer time scales than early exposure. (Finding 1.1-209) Several potential modes of chronic exposure exist, the most important of which are direct radiation from contaminated ground and ingestion of contaminated milk or crops. Doses from inhalation of resuspended material are included in the computation, but are small relative to doses from other pathways. (Finding 1.1-211)

Evacuation and sheltering reduce early exposure. Sheltering reduces the doses incurred from inhalation, external plume exposure, and ground contamination. Two chronic phase dose mitigation actions are interdiction and decontamination. Interdiction of both land and food were considered by Staff and licensees, and both parties agreed on the appropriate assumptions to be used. IPPSS assumed

that interdiction would not be applied to land contaminated at relatively low levels, that is, levels which would lead to only small increments in the risk of cancer fatality from other causes. Although one intervenor questioned whether effective decontamination of built-up areas is possible, Staff and licensees agreed upon decontamination assumptions. (Findings 1.1-212 through 214)

Health effects can be divided into two categories: early and latent effects. Early effects include fatalities and illnesses caused by threshold doses of radiation received in a short period of time, most within days to weeks after exposure. (Finding 1.1-216) The second category of health effects includes cancers, curable thyroid nodules, and genetic effects caused by lower doses received over a longer time. Virtually all of the cancer fatalities, thyroid cancer cases, and man-rem estimated in IPPSS result from low radiation doses for which estimates are based on extrapolation rather than direct experimental data. (Finding 1.1-217)

To minimize early fatalities, supportive medical treatment would be given to people receiving potentially lethal doses to the total bone marrow from early exposure (greater than about 175 rem). Only those persons receiving dose exposures exceeding 350 rem would need special facilities; thus, most persons needing supportive medical treament would not require such special facilities.

Moreover, medical treatment need not commence for 20 days to be effective. This allows sufficient time to identify the affected areas and to screen the population that will require special treatment. (Finding 1.1-219)

Intervenors questioned whether medical facilities or personnel would be available to administer supportive treatment to those who may suffer substantial radiation exposure. (Findings 1.1-220 through 1.1-223) The Board finds that emergency medical personnel will be responsible in the discharge of their duties, and that under virtually no set of circumstances would the number of people needing treatment exceed the capability of medical personnel and facilities.

4. Emergency Response Assumptions

An important component of consequence assessment is the emergency response assumptions. IPPSS consequence analysis for the 10 mile EPZ was based on the evacuation plan developed by consultants who prepared the evacuation time estimates for emergency planning purposes at Indian Point. (Finding 1.1-239) For areas within 10 miles, five different scenarios were developed to account for time-of-day and seasonal variations in evacuation and population. The appropriate population and associated evacuation data bases were selected based upon the start time for each accident scenario. (Id.)

Although intervenors questioned whether IPPSS' modeling of evacuation transit times was consistent with the emergency planning evacuation times, IPPSS used the most current data available at the time the analysis was performed. Delay time, the time between warning from the plant and the start of evacuation, is a key parameter in computing early health effects. IPPSS treated delay time probabilistically, reflecting the possibility that the entire evacuee population may be delayed longer than expected due to factors such as severe weather conditions, unavailability of transportation or difficulties in communication. For example, it was assumed that at least 30 percent of the population was delayed more than one hour in all scenarios. (Findings 1.1-239 through 1.1-241) Delay times of up to 13 hours were used in the IPPSS. The Board finds that the evacuation modeling assumptions employed in IPPSS are reasonable for the same reasons that it finds the evacuation time estimates proposed for emergency planning regulations. As we discussed under Commission Question 3 and Contention 3.3 below, these estimates were independently verified.

Staff's and licensees' emergency response assumptions differed principally because Staff chose three alternative sets of emergency response, rather than a probabilistic approach, to bound the analysis. The first set of parameters, designated by Staff as "Evac Reloc,"

assumes evacuation of the population within 10 miles. The second set of parameters, designated as "Early Reloc," reflects the hypothesis that the planned evacuation would not or could not take place, and that segments of the population within 10 miles would be relocated 8 hours after plume passage. (Findings 1.1-243 through 1.1-245)

Licensees' and Staff's consequence analysis both accounted for shielding effects associated with different types of protective actions. (Finding 1.1-242) While questioning the sheltering assumptions made in IPPSS, UCS/NYPIRG conceded that studies reported in the literature and the PRA Procedures Guide characterize the inhalation reduction factors used in IPPSS as conservative. The Board finds that accounting for shielding effects is reasonable and appropriate. (Id.)

The third set of emergency response parameters used by Staff, designated "Late Reloc," reflects the assumptions of an emergency response situation hampered by a severe external event which would seriously limit the ability to evacuate and reduce shielding protection. For these accidents, relocation of the public from highly contaminated areas 24 hours after plume passage was assumed. (Finding 1.1-246) This assumption is equivalent to no emergency response at all for 36 to 48 hours after accident initiation, and thus unreasonably overstates risk. Because people are unrealistically assumed to stand

outdoors for the entire period, they would not benefit from shielding, which lowers radiation dose. (Finding 1.1-247)

Intervenors argued that an 8 hour warning time should be used for late overpressurization failure of containment accidents, rather than the 11 hours used in IPPSS. Licensees testified that, based upon updated analyses, a late overpressurization release was not likely to occur until at least 24 hours after accident initiation, allowing 13 hours, a sufficient time for plant operators to realize that such a release could occur. Although UCS/NYPIRG argued that there would be a significant impact on the calculated consequences using an 8 hour warning time, which would reduce the amount of time available for implementation of protective actions, licensees' sensitivity analyses demonstrated that early fatalities within the evacuation zone could be prevented if evacuees begin moving no later than the start of the release and maintain a minimum speed of about three miles per hour. (Findings 1.1-248 and 1.1-250) At a few miles from the plant, evacuations at this speed could be initiated somewhat later without an increase in the risk of an early fatality. Furthermore, evacuation of any kind has only minor effects on latent fatalities. Intervenors conceded that utilizing both the licensees' and the Staff's emergency response assumptions would result in only a very small chance of early fatalities. (Finding 1.1-252)

5. Other Consequence Estimates

As we indicated above, the intervenors did not present any results based upon assessments of risk. Without presenting any probabilities, however, intervenors did present estimates of consequences. Intervenor witnesses Palenik/Beyea estimated large latent health effects in New York City from a postulated WASH-1400 PWR-2 release at Indian Point. These estimates far exceeded those of licensees or Staff. Although these witnesses did not know the actual frequency of the meteorological conditions which they assumed, it is clear from the testimony that it is less than 2% of the time. (Finding 1.1-317) Palenik/Beyea also made no adjustments for terrain, which would have a major effect at this particular site based upon the meteorological testimony in the record. (Finding 1.1-318) Palenik/Beyea used dose/health effects conversion factors which are inconsistent with the most recent Biological Effects of Ionizing Radiation (BEIR) Committee recommendation by factors ranging from 2 to 5. (Findings 1.1-230 to 231)

Consequence modeling was also advanced by intervenor witnesses Pisello/Piccioni, who postulated four scenarios. The witnesses used a simplified wedge model for modeling dispersion which they conceded was inappropriate

for calculating doses at the site, as their first scenario presumed. (Finding 1.1-334) They conceded that licensees' CRACIT model was preferable to the one they employed. (Finding 1.1-335) Another Pisello/Piccioni scenario modeled a plume traveling towards New York City. The witnesses applied no shielding factor although they recognized that one should be applied. (Finding 1.1-338) The third scenario modeled impacted the New York City upstate watershed, a scenario which was also examined by a Staff witness. Staff's analysis showed that if a release be assumed, the frequency of exceeding the maximum permissible concentration of strontium-90 in reservoir systems in the first year was 1.1 percent. (Finding 1.1-342) The fourth scenario postulated by Pisello/Piccioni predicted land interdiction at 50 miles from the site, which the Staff found to be of such low likelihood as to not warrant inclusion in its own analysis. (Finding 1.1-344) As did Palenik/Beyea, Pisello/Piccioni assumed dose/health effects conversion factors which substantially overstate effects compared to more traditionally used conversion factors such as BEIR. (Finding 1.1-345)

The Board concludes that because these witnesses employed oversimplified plume transport models, gave little or no weight to shielding factors, and used dose/health effects conversion factors which are contrary to the weight

of authority, their estimates of consequences are unpersuasive. We discuss probabilities associated with the Palenik/Beyea scenario in connection with Board Question 1.3 below.

CRITICAL ASSESSMENT OF IPPSS

After it was published in 1982 the IPPSS was sent to Sandia National Laboratory, which performed a critical review of the IPPSS under contract to the NRC Staff. The Sandia analysis was published in December 1982 as "Review and Evaluation of the Indian Point Probabilistic Safety Study," NUREG/CR-2934, SAND82-2929. The NRC Staff in turn reviewed both the IPPSS and Sandia's review of the IPPSS. Sandia and NRC reviews of the IPPSS consumed 5 to 10 man-years of effort. (Finding 1.1-21)

The Board specifically invited the parties to address a prior draft of the Sandia review as well as other materials which might bear upon the accuracy of the IPPSS. Reviewers from both the Staff and Sandia Laboratory gave testimony at the hearings on both IPPSS and their own alternative risk calculations for the Indian Point units. Intervenors did not present independent calculations of risk in accordance with the Commission's Order (13 NRC 1), however certain consequence modeling was presented and criticisms raised about the manner in which certain portions of the IPPSS were performed. We discuss the most significant issues regarding the accuracy of the IPPSS results here.

Some witnesses questioned the modeling of human error rates in the IPPSS, contending that it was

excessively difficult to model operator actions under conditions of stress as would pertain under potential accident conditions, as well as cognitive errors and errors of omission. Despite inherent shortcomings in any effort to accurately predict human behavior, the human error modeling in IPPSS represents a substantial extension of the human reliability analysis contained in earlier PRAs such as WASH-1400. (Finding 1.1-52) The IPPSS used the NRC's Human Reliability Handbook as its principal source of human error rate information, and modified the error probabilities given in the Handbook to account for plant-specific situations as recommended by the Handbook's author. (Id.) Human error rates were intentionally increased to account for stressful conditions where appropriate. Since Indian Point operators receive frequent and extensive simulator training under simulated accident conditions, it is much less likely that misdiagnosis of accident conditions would occur, and concomitantly, due to simulator training there is a convergence between actual operator behavior and the modeling of that behavior as performed for the IPPSS study. (Findings 1.1-53 and 1.1-56) The review of IPPSS performed by Sandia Laboratories determined that the likelihood of operator misdiagnosis was negligible (Id.), and a recalculation of four important IPPSS human error probability estimates by the Staff revealed no difference to the accuracy of the analysis. (Finding 1.1-58)

With any PRA there are inevitably issues as to whether all potential sequences which could lead to a serious accident at that particular plant have been included in the analysis. In the IPPSS, both industry-wide nuclear plant operating experience data and plant-specific data were collected. Although it is of course conceivable that some accident scenarios may have been omitted from the analysis, such omissions were unlikely because of the thoroughness of the analysis, and because the Sandia and Staff reviews were sufficiently comprehensive to identify most omissions or errors. Any such errors are likely to involve less important contributors to risk, since such a sequence would either have to bypass or breach containment to pose a risk to the public. Thus the contribution to the overall risk from unidentified scenarios is likely to be small. (Findings 1.1-61 through 1.1-63)

One potential accident initiator which was intentionally omitted from both the IPPSS and Staff risk analyses was sabotage. Sabotage was considered inappropriate and incompatible with publication in a publicly available document. A probabilistic evaluation of sabotage would necessarily require a detailed discussion of both possible sabotage scenarios and plant security arrangements. (Finding 1.1-70) The Board notes that the NRC's recently published PRA Procedures Guide, NUREG/CR-2300, states, at 10-1, that in performing nuclear

power plant PRAs "events like sabotage and war are not included." The Commission's recent Policy Statement on Safety Goals for the Operation of Nuclear Power Plants similarly excludes sabotage from its reach. While not strictly related to PRA, by their nature the safety goals imply quantitative assessments of risk, yet the Commission has expressly excluded sabotage from safety goal consideration, stating that:

> "The possible effects of sabotage or diversion of nuclear material are also not presently included in the safety goal. At present there is no basis on which to provide a measure of risk on these matters." 48 Fed. Reg. 10773 (March 14, 1983)

Thus sabotage is normally excluded from all clear plant PRAs, with the encouragement of the NRC. The Board finds that the licensees and Staff were reasonable in their decisions to exclude sabotage from their respective assessments. Moreover, since there is nothing in the record to suggest that the Indian Point units are any more susceptible to sabotage than other nuclear plants, the omission of sabotage from consideration in this and other PRAs as well can have no net effect on the comparison of the risk at Indian Point with other plants, which the Commission indicated was its "primary concern," 13 NRC at 5.

A further issue which was raised regarding the IPPSS related to the treatment of common cause failures, which refers to multiple failures of safety equipment

resulting from a common cause. Among the potentially most significant common cause failures are external events such as hurricanes and earthquakes, which could knock out power supplies, cooling systems or other common support equipment simultaneously. Since IPPSS was one of but a very few PRAs to attempt assessment of external events, it can only represent an improvement in the evaluation of externally caused common failures. The testimony before us indicated that common cause failures were modeled in the IPPSS plant event trees and system fault trees, and that the Staff's independent analysis determined that the more important common cause failures had been modeled. (Finding 1.1-66) New industry data concerning common cause failures which became available after publication of IPPSS were included in the Staff analysis, and while no attempt was made to ascertain their incremental effect, there was no indication that it was substantial. (Finding 1.1-68) Witnesses who identified common cause failures as an area of concern were not able to point out any particular failure modes which had not been identified by the IPPSS, Staff or Sandia Laboratories assessments. (Finding 1.1-69)

In the event of most of the serious accidents which could occur at a nuclear power plant, the ability of the containment structure to function as intended to contain the accident would be the major factor in whether or not there would be any offsite consequences. Contain-

ment capability is a matter of structural strength, since the challenge to the containment would be from increased pressures created as a result of the accident. The ultimate failure pressure is the pressure at which the containment would in fact rupture resulting in a release of radioactive materials through the breach to the environment. (Finding 1.1-134) Because of differences in materials and construction, failure pressure may vary substantially from one plant to another. If pressure within a containment were to become great enough, there would start to be yielding of reinforcing material at the most limiting location within the containment structure. (Finding 1.1-134) Beyond this point the containment would experience plastic deformation, during which the containment would be severely stressed but would not rupture. (Id.) The testimony indicates that the steel liners inside the Indian Point containments are capable of extensive plastic deformation prior to failure. (Finding 1.1-145) At pressures beyond the range of plastic deformation, the containment would actually breach.

The record before us reveals considerable uncertainty about the pressure level at which there would be an actual breach or rupture of the Indian Point containments. (Finding 1.1-140) In order to overcome this uncertainty about breach pressure, both the licensees and the Staff assumed for purposes of their respective accident

analyses that containment breach would occur not at ultimate strength of the structure, but at that point prior to plastic deformation where yielding of structural members would first start to occur. (Id.) This was a conservative assumption to make, in that it results in breaches (and releases) occurring earlier and with greater frequency than would be the case had ultimate strength been used in modeling accidents as they were assumed to unfold. The analysts did this because unlike ultimate strength, there is much less uncertainty surrounding the point of onset of yielding of a containment structure. (Findings 1.1-139 through 1.1-143)

Both the licensees' and Staff's analysis determined that the point of onset of yielding of both of the Indian Point containments was approximately 141 psia, which corresponds to 126 psig. (Finding 1.1-135) These determinations were independently arrived at through detailed calculations using standard and well-understood structural engineering principles which accurately predict the behavior of steel reinforced concrete structures at their elastic limits. Intervenor witnesses performed no independent analysis of containment failure pressure, nor reviewed the methodology used by the Staff and licensees to determine the point of onset of yielding. Their opposition to the use of 141 psia as the assumed failure pressure of the Indian Point containments was founded upon the absence

of scale model tests to failure, and claimed uncertainty surrounding the calculation of ultimate failure pressures. (Finding 1.1-143) However, the sources cited in support of the supposed uncertainty do not relate to onset of yielding, which would assuredly occur prior to breach, and which based on the uncontroverted record before us is capable of determination through structural analysis. (Finding 1.1-141)

The Board finds that the licensees' and Staff's use of the point of onset of yielding as the assumed ultimate failure pressure of the Indian Point containments for purposes of modeling accident sequences was a conservative and reasonable assumption that overcame concerns over the difficulty of determining an actual failure pressure. We find that the determination of 141 psia as the point of onset of yielding of the Indian Point containments is fully supported in the record by the two independent analyses, each of which produced an identical value.

SUMMARY OF RISK RESULTS

In this proceeding both the licensees and the Staff presented probabilistic risk assessment (PRA) results for the Indian Point plant. The licensees' results were presented principally as a family of risk curves, which represent a complete display of risk including uncertainty. Such risk curves provide the most information regarding risk. Because the Staff did not quantify uncertainty, their risk curves are presented as point estimates. Staff acknowledged that while the uncertainty bands accompanying their results are substantial, it is more likely that they have overestimated rather than underestimated risk. (Findings 1.1-346 and 347) In the hearings the intervenors did not present any results based upon an assessment of risk. They did put forth selected consequence estimates disassocated from any probabilities for the Indian Point units.

1. Early Fatality Risk

If a severe accident were to occur at Indian Point, the probability is extremely low that sufficient radioactivity could be released to cause potentially lethal exposures. Employing source terms which the Staff and licensees both agree are conservative (Findings 1.1-308 and 1.1-312), the licensees' median frequency (50 percent

confidence level) estimate of an accident at Indian Point Unit 2 which would cause one or more early fatalities* is once in 110,000,000 reactor years. The median frequency for accidents causing 100 or more early fatalities is once in 500,000,000 reactor years, and for 1000 or more early fatalities, the licensees' median frequency is once in 40,000,000,000 reactor years. Similarly, for accidents at Indian Point Unit 3, the licensees' median frequency (50 percent confidence level) of one or more early fatalities is once in about 140,000,000 reactor years, the median of 100 or more early fatalities is once in about 700,000,000 reactor years, and the median frequency for 1,000 or more early fatalities is once in about 45,000,000,000 reactor years. (Findings 1.1-349 through 1.1-351)

The Staff presented a slightly higher estimate of early fatality risk from accidents at Indian Point. Their "best estimate" early fatality risk curve, which employed source terms acknowledged to be overly conservative and assumed very long delays in evacuation for earthquake and hurricane initiated accidents, showed for accidents at Indian Point Unit 2 one or more early fatalities occurring

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* An early fatality is defined as a death attributable to radiological causes occurreing within 60 days of exposure.

with a frequency of about once in 200,000 years, 100 or more early fatalities occurring about once in 400,000 years, and 1,000 or more such fatalities occurring about once in 600,000 years. (Finding 1.1-353) On a similar basis, for accidents at Indian Point Unit 3 the Staff's best estimate early fatality curve shows a frequency of about once in about 500,000 reactor years for one or more, once in about 1,000,000 reactor years for 100 or more, and once in about 1,670,000 reactor years for 1,000 or more. (Finding 1.1-354)

The licensees and Staff also estimated the early fatality risk to individuals living in the vicinity of Indian Point. Licensees estimated the early fatality risk to individuals within one mile of Indian Point to be about once in 140,000,000 per reactor year and about once in 160,000,000 per reactor year for Units 2 and 3 respectively. (Finding 1.1-356) Licensees also estimated that more than 95 percent of the early fatality risk from all core melt accidents, even assuming that the public would take no protective measures, would be confined to within 4 miles of the site. Less than 0.2 percent of the early fatality risk is beyond 10 miles. (Finding 1.1-370)

The licensees found that the dominant contributor to early fatality risk at Indian Point is the interfacing systems LOCA, which accounts for 97 percent of the early fatality risk. (Finding 1.1-369) This accident, which has

a median frequency of occurrence of once in 25,000,000 reactor years according to licensees and once in 2,500,000 reactor years according to Staff, is initiated by failure of two valves which isolate the residual heat removal (RHR) system from the primary coolant system. Since the primary system is usually at much higher pressure, the failure of the valves can result in an overpressurization of the RHR system and loss of primary water. Although the interfacing systems LOCA is rare, nevertheless this event still dominates the early fatality risk because all other contributors are even lower. (Id.)

2. Evaluating the Differences in Licensees' and Staff's Estimates

Differences in assumptions about the emergency response actions which would be taken following a serious accident can be very important in estimating early fatalities, and largely explain the differences in licensees' and Staff's frequencies. IPPSS analyzed evacuation delays probabilistically for all accident scenarios to account for the possibility that emergency response could be slower than expected for any reason including adverse weather and road conditions. Thus, delay times as long as 13 hours were used for certain segments of the population. Staff, on the other hand, assumed that if an accident were caused by a severe earthquake, no one

would evacuate or take shelter for 24 hours after plume passage. Since containment failure and release are not expected to occur until at least 12 hours and more likely 24 hours of the initiating event, Staff's assumption is the equivalent to assuming no emergency response at all for 36 to 48 hours after accident initiation. Because Staff's assumption would have people stand outdoor for the entire period, they would not benefit from shielding, which would lower radiation dose. (Finding 1.1-360)

In IPPSS, the gradual overpressurization of containment accident was found not to be a contributor to early fatality risk primarily because with more reasonable emergency response assumptions there was sufficient time for effective evacuation. (Finding 1.1-362) When Staff evaluated the same accident sequence assuming that evacuation would occur, it too found that there would be no early fatalities. (Id.) Intervenors' witness agrees that Staff's "no evacuation" assumption was a worst case analysis. (Finding 1.1-364) The Board finds that the Staff's assumption of no emergency response for 24 hours after plume passage from earthquake and hurricane initiated releases is unduly conservative and overstates early fatality risk.

Another reason for the differences between licensees' and Staff's early fatality risk estimates is the treatment of the accident caused by a direct seismic

failure of the containment building. The Staff assement included contributions to the early fatality risk from an earthquake-initiated collapse of the containment building. The licensees testified that the Indian Point containment buildings are strong enough to withstand the largest earthquake that could occur in the vicinity, based upon the analysis which we referred to above in discussing external events, and that there is therefore no contribution to the early fatality risk from this release category. At the time of the hearings the Staff had not yet evaluated the licensees' testimony that there can be no direct seismic failure of the Indian Point containments, but agreed that if the licensees' reanalysis was correct the overall risk would be reduced. Elimination of the late overpressurization accident as a contributor to early fatality risk and the elimination altogether of the direct seismic failure of containment reduces the Staff's estimate of overall risk and, the mean early fatality risk by factors of 25 and 6 for Indian Point Units 2 and 3, respectively. (Finding 1.1-366)

The Staff also indicated that the source term they used may have been significantly overestimated. The impact of lesser values of source terms would be substantially lower estimates of health effects, particularly early fatalities. Anticipated reductions in the source terms would significantly reduce the early fatality risk. The

licensees testified that a reduction of a factor of 15 in the IPPSS source term would effectively eliminate the early fatality risk, assuming evacuation as modeled in the IPPSS. (Finding 1.1-367)

3. Latent Cancer Fatality Risk

Licensees and Staff each reached approximately the same conclusion with respect to their estimates of the risk of latent cancer fatalities from serious accidents at Indian Point. (Finding 1.1-371) Again using source terms which Staff and licensees acknowledge are conservative, (Findings 1.1-308 and 1.1-312), both agree that the risk of latent cancer fatalities is very small, and would represent only a slight, and probably not statistically significant, increase above the naturally occurring cancer rate. (Finding 1.1-377) Licensees estimated that the median frequency of accidents resulting in an increase in the background cancer rate exceeding one-one hundredth (0.01) of one percent is once in 23,000 reactor years for Unit 2 and once in 170,000 years for Unit 3. (Finding 1.1-377) The median frequencies of exceeding these rates by one-tenth (0.1) of one percent are once in 50,000 and 330,000 reactor years respectively, and the median frequencies of exceeding them by one percent are once in 10,000,000 and 100,000,000 reactor years respectively. (Id.) The licensees' estimates of latent cancer fatality

risk to an individual within one mile of Indian Point is one in 16,000,000 and 62,000,000 per reactor year for accidents at Indian Point Units 2 and 3 respectively. (Finding 1.1-378) The Staff presented estimates of the individual risk of delayed (latent) cancer fatalities on a per site basis as a function of distance. The Staff's estimate of individual risk is low, ranging from about one in 1,200,000 per reactor year to about one in 300,000,000 per reactor year. The latent cancer fatality risk to an individual within one mile of the site is about one in 1,700,000 per reactor year. (Id.)

The dominant contributor to latent fatality risk at Indian Point is a radioactive release resulting from a gradual overpressurization of containment. Licensees estimated that the median frequency of containment overpressurization failures at Indian Point Units 2 and 3 is 30,000 and 170,000 reactor years, respectively. Staff's calculated frequency for this release is about once in 29,000 and once in 57,000 reactor years for Indian Point 2 and 3, respectively. (Finding 1.1-379)

External initiating events are the major contributor to latent fatality risk from accidents at Indian Point Unit 2. The largest of these contributors is severe wind (hurricane), far beyond the highest level ever recorded in the Indian Point area. Such a hurricane was assumed to cause loss of AC power which, assuming no

recovery, would lead to a late core melt followed by containment overpressurization. The mean frequency of this scenario is about once in 37,000 reactor years. (Finding 1.1-380) During the course of the hearings a hurricane watch was put in place at Unit 2, and procedures initiated whereby the unit would be shut down if a severe hurricane were to threaten the site. With these procedures, Staff testified that the frequency of this scenario is substantially lower. The second largest contributing scenario at Unit 2 is initiated by a tornado, which would cause simultaneous loss of onsite and offsite power and other safety-related equipment. The mean frequency of this scenario is about once in 63,000 reactor years. The next largest contributor is a seismically-induced complete loss of power scenario which, assuming no recovery, would lead to a late core melt and delayed overpressurization of containment. The mean frequency of this scenario is about once in 145,000 reactor years. (Id.)

Fires originating in either the switchgear or cable spreading rooms are the main contributor to latent cancer fatality risk from accidents at Indian Point Unit 3. Such fires could lead to a loss of AC power and eventual containment overpressurization. The mean frequency of fires leading to latent fatalities is once in 140,000 reactor years. The next largest contributor to latent fatalities is a large seismic event causing loss of

control power or loss of AC power. This accident scenario has a mean frequency of once in 400,000 reactor years. (Finding 1.1-381)

4. Other Health Indices

With regard to health indices other than early and latent fatality risk, the licensees presented societal risk curves for non-fatal radiation illnesses and thyroid cancers resulting from accidents at Indian Point Units 2 and 3, respectively. The median frequency of accidents at Indian Point Unit 2 causing one or more non-fatal radiation illnesses is once in 2,000,000 reactor years; for 100 or more such illnesses, once in 20,000,000 reactor years; and once in 750,000,000 reactor years for 1,000 or more illnesses. The Staff's point estimate early injury (radiation illness) risk curve for Unit 2 shows the frequency of one or more radiation illnesses to be about once in 25,000 reactor years, 100 or more to be about once in 33,000 reactor years and, the frequency of 1,000 or more to be about once in 50,000 reactor years. (Finding 1.1-383)

The median frequency of one or more non-fatal radiation illnesses at Indian Point Unit 3 is once in 6,400,000 reactor years, the median frequency of 100 or more illnesses is once in 45,000,000 reactor years, and the median frequency of 1,000 or more such illnesses is once in

950,000,000 reactor years. According to the Staff's Unit 3 point estimate early injury (radiation illness) risk curve, the frequency of one or more radiation illnesses is once in about 50,000 reactor years, the frequency of 100 more radiation illnesses is once in about 100,000 reactor years, and the frequency of 1,000 or more radiation illnesses is once in about 160,000 reactor years. (Finding 1.1-384)

The licensees estimated that average risk of radiation illness to an individual within one mile of the site from accidents at Indian Point Units 2 and 3 to be about one in 14,000,000 per reactor year and one in 48,000,000 per reactor year, respectively. The Staff presented the individual risk of early injury (radiation illness) on a per site basis as a function of distance. The risk is low, ranging from about one in 1,100,000 to 1,250,000,000 per reactor year. The Staff's estimate of risk of early injury caused by Indian Point to an individual one mile from the site is about once in 1,100,000 per year on a per site basis. (Finding 1.1-385)

The licensees' estimate of the median frequency of accidents at Indian Point Unit 2 causing one or more thyroid cancers is about once in 8,500 reactor years, and about once in 54,000 reactor years for 1,000 or more. Thyroid cancers are curable in approximately 90 percent of the cases. Licensees' estimate of the median frequency of one or more thyroid cancers from accidents at Indian Point

3 is once in 25,000 reactor years, and of 1,000 or more it is once in 350,000 reactor years. (Findings 1.1-386 and 387)

5. Offsite Property Damage

With regard to offsite property damage, the licensees constructed a family of risk curves expressed in 1982 dollars to display the property damage that may result from reactor accidents at Indian Point Units 2 and 3. Licensees' estimate of the median frequency of any offsite property damage caused by an accident at Indian Point Units 2 and 3 is once in 6,700 and once in 17,000 reactor years respectively. (Findings 1.1-389 and 390) Staff presented a point estimate property damage curve in 1981 dollars which shows the frequency of any offsite property damage from accidents at Indian Point 2 and 3 to be once in 3,300 and 10,000 reactor years respectively. The Staff's estimate of the annual average property damage risk for the two unit site in 1981 dollars amounts to $450,000 per year. Although a realistic estimate of the background economic loss rate within 50 miles of the site has not been developed, the Staff concluded that a wide variety of accidental hazards would pose economic loss rates well in excess of this rate. (Findings 1.1-391 and 1-1-393)

SOURCE TERMS

An important step in analyzing the likely consequences to the public from postulated reactor accidents is determining the quantity and chemical form of the radionuclides which might be released to the environment. This determination is commonly referred to as the source term. Overly conservative estimates of the magnitude of the source term released to the environment (i.e., overstatements of the quantities of radionuclides released) lead to inaccurate and misleading overestimates of accident consequences, which in turn could encourage inappropriate decision-making in response to accidents. (Finding 1.1-277) In addition, the use of conservative source terms would overstate the value of mitigative devices such as those which have been considered in connection with Commission Question 2 in this proceeding. (Id.)

The earliest efforts at reactor accident consequence modeling such as the Reactor Safety Study (or WASH-1400) and its predecessors made very conservative assumptions regarding the composition and amount of fission products which could escape from the core of the containment, and subsequently to the environment, in the event of a severe accident. In particular, these studies assumed that radioactive materials released from the core

could pass through the primary system, the containment, and then out to the environment without any significant reduction. The early studies also assumed that two of the most important radionuclides from the standpoint of early effects -- cesium and iodine -- would remain in their elemental forms and pass without attenuation from the reactor vessel to the primary system and to the containment. Since the IPPSS used basically the same assumptions regarding radionuclide behavior as the Reactor Safety Study, the results in IPPSS for potentially severe accident sequences show very little change from the earlier studies. (Finding 1.1-278)

Recently there has been increased attention given to researching how radionuclides would actually behave following a serious accident, and researchers have examined data from TMI and other accidents for clues as to source term behavior. (Finding 1.1-281) Source term research is ongoing at present both within the NRC and industry. However, the results of research completed to date call into question the basic assumptions underlying the source terms used in IPPSS and the earlier studies. (Findings 1.1-284 through 1.1-288) This recent work has suggested that there are many important physical and chemical processes which had previously been neglected which would act to retain certain fission products within the damaged facility, thus preventing their release to the environment.

Both the licensees and Staff agree upon a number of processes such as solution, sedimentation, and agglomeration as being important to predicting fission product behavior. (Finding 1.1-280) Both the Staff and licensees also agree that based upon analyses of releases during TMI, the prior assumptions that cesium and iodine would remain in their elemental, volatile form following release from the core are incorrect. (Finding 1.1-281) Instead, the expected form of iodine will be cesium iodide and not elemental iodine. Since cesium iodide is highly water soluble, once in solution it would not become airborne and available for release. Since iodine accounts for half of the potential early fatality risk, the formation of cesium iodide could have a significant effect on offsite effects. (Id.)

In the hearings, licensees offered testimony regarding data from TMI and a number of other actual accidents as well as experimental data which tended to confirm the physical and chemical processes which would reduce the amount of radionuclides that could be released. Based upon the prior research and data available, the licensees presented revised estimates of more realistic but still conservative source terms as compared to those in WASH-1400 and IPPSS. The factors relied upon by licensees' experts are the physical and chemical properties of the radioactive materials, and do not depend upon operator

action or equipment functioning. Licensees presented revised source terms for the two release categories found in IPPSS to dominate risk at the Indian Point units, the interfacing systems LOCA and the slow overpressurization accidents. (Finding 1.1-294) In developing these source terms, attenuation for each isotope through various barriers within specific Indian Point plant systems was considered and a decontamination factor for each mechanism assigned. The revised values presented by licensees' experts are the upper limits on the amount of fission products which they believe could escape from the plant. (Finding 1.1-296)

Four measures of accident risk were reevaluated utilizing the revised source terms. The results indicate that even when making the unrealistic assumption that no one evacuates or takes shelter for 24 hours, no early fatalities would result from any serious accident at Indian Point. Licensees' more realistic reanalysis results in a reduction in the mean latent fatalities per year by a factor of approximately seven, man rem is reduced by a factor of 10, and property damage is reduced by a factor of 17. This would reduce the health effects consequences of even a "worst case" accident to those approximating other large-scale industrial accidents. (Findings 1.1-302 and 1.1-304 through 1.1-306)

The Staff was questioned about licensees' reanalysis of risk using revised source term values. Staff stated that it had no data or information which were inconsistent with the licensees' reanalysis, and while it could not confirm licensees' revised source terms at this time, it could not state that anything in licensees' reanalysis was in error. (Finding 1.1-308) Staff did agree with licensees that the accident source term methodology used in WASH-1400 results in overestimates of the quantity of fission products released. (Id.) Staff also testified that the core heatup and aerosol computer codes used in the Staff's analysis of IPPSS result in an overstatement of the source term. (Finding 1.1-310)

The Board concludes that the consequence data developed by licensees' witnesses using their revised source terms represent the most realistic and thus most appropriate information before us upon which we may rely in determining the expected effects of serious accidents at the Indian Point plant. We cannot say that the licensees' source terms are correct and that some other source terms are incorrect. But what we can and do find is that the source terms and accompanying consequence analysis presented by licensees' witnesses are less likely to misrepresent radionuclide behavior, and thus offsite releases, than are others put forth in the hearings. Based on the evidence and cross-examination, the source terms

employed in the Staff analysis and IPPSS are conservative and result in overestimates of the already low risk from operation of the Indian Point facilities.

We believe that it is the Board's obligation to select the most likely result from the available alternatives and conclude that we do so by accepting that of licensees' experts. It is undisputed that competing source terms such as those used in WASH-1400 and IPPSS contain unrealistic assumptions and ignore physical and chemical processes, such as the formation of cesium iodide, which are now known to be correct. Staff's witness testified that it was his personal prognosis that there will be a reduction in overall source term releases on the order of a factor of ten. (Finding 1.1-312) Staff's overall position seems to be that until research for the precise quantification of source terms is available, reductions from WASH-1400 values are inappropriate. This may be a desirable position from the standpoint of overall Commission policy, but we find that we must employ a best estimate for purposes of responding to the Commission's Questions here. Staff's witness testified that an understanding of the biases introduced by current WASH-1400-based source terms was use necessary to make informed regulatory judgments (Tr. 12597-602), and the Board believes that it has an obligation to remove such biases in its own decision when a choice between two

alternatives is presented. The source term which we select as most likely would dramatically reduce the consequences of even the most severe reactor accidents.

BOARD'S CONCLUSION WITH RESPECT TO COMMISSION QUESTION ONE, CONTENTION 1.1 and BOARD QUESTION 1.1

The Board finds that the expressions of risk for the Indian Point units as contained in the licensees' testimony provide a reasonable and accurate basis for assessing the risk of serious accidents at those units under assumptions of radionuclide behavior (source teams) which have traditionally been used in nuclear power plant probabilistic risk assessments. The Indian Point Probabilistic Safety Study which forms the basis for most of licensees' testimony is a competently performed PRA which contains several important and pioneering advances in the art, including quantification of uncertainty, analysis of external initiating events, data accumulation and assemblage, and realistic consequence modeling. The Board accordingly finds that the risk curves set forth in Appendices A and B to our Findings constitute reasonable representations of the risk associated with the operation of the Indian Point units as they are presently configured.

Differences between the presentations of Indian Point risk contained in licensees' testimony and those of the NRC Staff are primarily attributable to relatively

minor differences in data or methodology and do not materially differ from those of the licensees given the ranges of uncertainty contained in all of the results. After weighing all of the evidence, we find that the reasonableness of licensees' and Staff's expressions of risk are not materially affected by the criticisms which were advanced by intervenors. Accident consequence estimates advanced by Palenik/Beyea and Pisello/Piccioni are of substantially reduced value because of the absence of probability estimates, although some of the assumptions in those analyses, such as dose/health effects relationships, are against the weight of authority.

The Board concurs with the assessment of both the Staff and licensees that the risk posed by the Indian Point units for every health effect studied is extremely low.

The Board further finds that the sources term assumptions currently in use in nuclear power plant probabilistic risk assessments are overstated by a significant but indeterminate amount.

THE BEARING OF THE PRECURSORS REPORT TO INDIAN POINT RISK

In 1982 a document was published by NRC entitled "Precursors to Potential Service Core Damage Accidents: 1969-1979, A Status Report," NUREG/CR-2497 (1982). That report attempts to calculate the frequency of severe core damage (SCD) accidents at nuclear power reactors in the United States between 1969 and 1979. A severe core damage frequency range of from 1.7×10^{-3} to 4.6×10^{-3} per reactor year was derived. The Reactor Safety Study (RSS) calculated a core melt frequency (synonymous with a SCD frequency) of 5×10^{-5}, a factor of nearly 100 less than the NUREG estimate. By our Board Question 1.2 we asked the parties to address the bearing, if any, that the NUREG report has on the reliability of IPPSS.

The probabilities of core melt found in the IPPSS are lower than the NUREG's estimate by approximately a factor of 10. Because the NUREG estimates an average SCD (i.e., core melt) frequency industry-wide, and the IPPSS estimates the Indian Point core melt frequency specifically, neither study can contradict nor invalidate the other. (Finding 1.2-19) Thus, the NUREG does not confirm or discredit the IPPSS results. Moreover, the IPPSS included the data and the incidents which were reported and analyzed in the NUREG. (Finding 1.2-13)

The NUREG also calculated SCD frequency by weighting "Near misses," i.e., events which could have become SCD accidents but did not, according to the probability that they would have become SCD accidents. These events were then combined and considered to equal one SCD. The TMI accident was combined with the derived SCD to yield a frequency of twice in 432 reactor years. Three events -- TMI, Browns Ferry, and Rancho Seco -- make up 82 percent of the total SCD frequency. (Finding 1.2-8) The NUREG has been severely criticized for this method of calculation. The Institute of Nuclear Power Operation found that the NUREG calculated the near misses 30 times too high, and that it failed to take into account improvements implemented in response to the lessons learned from these occurrences. (Finding 1.2-6)

The Staff presented testimony stating that the NUREG SCD frequency estimate is "unduly pessimistic" with respect to predicting the frequency of these types of accidents at Indian Point. The Indian Point plant design is fundamentally different from TMI in ways that decrease the likelihood of TMI-type accidents occurring at Indian Point. (Finding 1.2-18) Even if a TMI-type accident were to occur at Indian Point, the accident would likely be well contained so long as there is minimal containment heat removal. (Finding 1.2-12)

With respect to the remaining precursors, which constitute 18 percent of the calculated SCD frequency, almost all involve a loss of main feedwater or a loss of offsite power, both of which the IPPSS thoroughly analyzes. The NUREG assessment is unduly pessimistic as applied to Indian Point because the NUREG failed to account for successful core cooling utilizing the emergency core cooling system which could mitigate loss of main feedwater accidents at Indian Point. Nor did the NUREG consider the possibility that one of three gas turbines at the Buchanan Substation could be used to replace offsite power. (Finding 1.2-17)

Conclusion

The Board finds that NUREG/CR-2497 does not affect the reliability of the results of IPPSS because IPPSS takes into account the data analyzed in the NUREG. By attempting to calculate a core melt frequency for all plants the NUREG gives insufficient weight to the plant design of the Indian Point units.

PALENIK/BEYEA CONSEQUENCE PROBABILITIES

Testimony given early in the proceeding by Mr. Palenik and Dr. Beyea did not discuss the probabilities associated with certain consequence modeling which they performed. By our Board Question 1.3 we asked the parties

to address those probabilities. Staff presented testimony stating that the Beyea/Palenik estimates were not intended to assess risk in regard to probability, but instead to analyze potential accident conditions in response to the emergency planning issues involved in this proceeding. Accordingly, any probability estimate has little or no meaning. (Finding 1.3-11)

The licensees presented testimony that the probability of the occurrence of the meteorological conditions necessary for the Palenik/Beyea "average" distribution of the material released during an accident at Indian Point is approximately 0.15 for the interfacing systems LOCA and 0.083 for slow overpressurization accidents. The Beyea-designated "average scenario" yields a total frequency of once in 180,000 reactor years when the specific sequence frequencies are multiplied by their corresponding probabilities, and contributions from Indian Point Units 2 and 3 are conservatively added. The frequency of once in 180,000 years applies to the lower range of the Beyea/Palenik consequences: 1,345 latent fatalities. (Finding 1.3-13) Testimony was presented by licensees which showed that the higher Palenik/Beyea estimates are much less likely to occur because of significant conservative assumptions which, when combined with one another, lead to an unreasonable overestimate of risk. (Finding 1.3-14) Licensees and Staff agreed that

the Beyea/Palenik assumptions regarding health effects conversion factors and effectiveness of protective responses are not appropriate. (Finding 1.3-13 and 14) Because the likelihood of rain is low, the probability of Palenik/Beyea's precipitation scenario decreases to approximately once in 1,800,000 reactor years. (Id.) Licensees calculated the frequency of achieving the radiological conditions calculated by Palenik/Beyea for thyroid module cases as once in 20,000,000 reactor years for Units 2 and 3 combined. (Finding 1.3-15) The licensees' analysis employed IPPSS source terms which the Board has found to be unduly pessimistic in that they overstate risk. (Finding 1.3-16)

Conclusion

The Board finds that probabilities associated with the Palenik/Beyea consequences are extremely low. Even for the lower end of the range of latent cancer fatalities calculated by Palenik/Beyea, the probability is in the range of once in 100,000 to once in 200,000 years. The probability of ever reaching the maximum number of latent fatalities calculated by Palenik/Beyea would be much lower than that, if possible at all, because of the extremely conservative assumptions employed in their analysis. Anticipated reductions in the magnitude of radiological source terms would lower even further the probability of

occurrence of the radiological conditions necessary to arrive at the Palenik/Beyea consequence estimates.

THE RISK ASSOCIATED WITH PRESSURIZED THERMAL SHOCK

Pressurized thermal shock (PTS) refers to an accidental rapid cooldown of the water in a reactor vessel while the system pressure is maintained at a high level. A rapid cooldown of the inner surface of a reactor vessel causes thermal stresses in the vessel wall. These stresses combined with pressure stresses may cause preexisting small cracks to grow larger. Reactor vessel integrity could be lost by crack extension through the vessel wall. By Board Question 1.4 we invited the parties to assess the risk of PTS at the Indian Point units.

The NRC Staff is establishing a new rule which sets forth screening criteria based on a material property which is related to the fracture resistance of the reactor vessel. This material property is called the Reference Temperature for the Nil Ductility Transition (^{RT}NDT). (Finding 1.4-4) PTS risk is acceptable for those plants which have an ^{RT}NDT below the screening criteria. The Staff has selected values of 270°F for axial weld materials in the reactor vessel and 300°F for the circumferential weld material. (Finding 1.4-7)

The Licensees both submitted analyses comparing their plant design with a generic Westinghouse

probabilistic risk assessment of PTS. These analyses examined whether any plant-specific features at Indian Point would make PTS more or less likely than that determined in the generic study. As of December 1981, Indian Point Unit 2 was at an ^{RT}NDT value of 189°F, compared to the screening criterion for the axially-oriented weld of 270°F. (Finding 1.4-12) Based on the December 1981 ^{RT}NDT value of 189°F at Indian Point Unit 2, the unit is in the range of ten years behind the plants of greatest concern with regard to the day at which they will exceed the NRC Staff screening criterion for sensitivity to PTS. (Id.)

In order to assure that the screening criterion will not be reached before the end of the service life of Indian Point Unit 2, the licensee has received encouragement from the Staff to take actions to reduce fast neutron flux at the pressure vessel wall to slow the rate of increase of ^{RT}NDT. (Finding 1.4-17) One method of achieving such a flux reduction is the implementation of a low-leakage fuel loading pattern (LLLP). (Finding 1.4-18)

After studying several possible low-leakage fuel loading arrangements, Con Edison directed Westinghouse to incorporate an LLLP into the core design for fuel Cycle 6, which commenced operation in January 1983. Con Edison presently intends to continue LLLP throughout the remaining service life of the plant. Based upon the Indian Point 2

specific analysis performed by Westinghouse, the LLLP design incorporated in Cycle 6 will reduce fast neutron flux at the point of peak vessel wall exposure by 44% compared to a standard fuel loading pattern. The RT_{NDT} values at the end of life of Unit 2, utilizing the LLLP design, will be well below the NRC's screening criteria for the circumferential and axial welds.

The risk of PTS was also probabilistically evaluated in IPPSS. The IPPSS evaluated the frequency of a reactor vessel rupture large enough to exceed the capability of the ECCS. This evaluation yielded a mean frequency of once in 3,400,000 reactor years for all types of vessel failure, which included those failures identified as PTS events. (Finding 1.4-14)

Conclusion

The Board finds that the risk to public health and safety from an accident at the Indian Point units initiated by PTS is substantially less than those events which dominate risk, and will remain so for the balance of the service lives of the units.

IMPROVEMENTS IN THE LEVEL OF SAFETY AT THE INDIAN POINT UNITS

The second Commission question to the Board in this proceeding asked the parties to take the "base case" safety level of the Indian Point units as determined in Commission Question 1 and perturb that level of safety by assuming various additional safety measures were in place, and assessing the reduction in risk, if any, that would be associated with each such measure. The initial safety measure to be evaluated was a collection of operational, maintenance and training procedures exceeding the regular requirements of Commission regulations which were imposed upon the Indian Point licensees by February 11, 1980 Orders of the Director of the Division of Nuclear Reactor Regulation. First, in the area of conduct of operations, surveillance testing and maintenance, the Orders were intended to make transients less frequent, selected safety systems more reliable, and emergency response more comprehensive. Second, in the area of staffing, the Orders required that two rather than one Senior Reactor Operators be on each shift in each unit, and that the pool of safety consultants available to each licensee be expanded. Third, the Orders required that control room operators and onsite emergency response teams undergo expanded and accelerated training for severe reactor accidents. Fourth, engineering

margins related to limiting core temperature excursions during large loss of coolant accidents (LOCA) were increased. Finally, additional studies of the susceptibility of the plants to severe accidents were ordered to increase licensee understanding of possible accident modes and to explore opportunities for further risk reduction. The Director's Orders provided for a phased completion of the required tasks, and testimony in the hearings established that the licensees either had or were in the process of complying with each item as required. (Finding 2.0-4)

While there is no doubt that the Director's Orders resulted in an enhancement of safety levels at the Indian Point units, this cannot be expressed in quantitative terms. This is in large part due to the fact that the IPPSS analysis generally modeled the plant as it existed when IPPSS was published in 1982, and did not attempt a "before and after" analysis of the Director's Order measures. Furthermore, since the plant-specific data base for the Indian Point units employed in IPPSS relied on the entire operating history of the units, it was not possible to use PRA techniques to isolate the risk impact of just the staffing and training changes attributable to the Director's Orders. (Finding 2.0-5) The Staff did quantify the risk reduction associated with Director's Orders provisions relating to safety system design, surveillance

testing and changes in technical specifications, estimating that they would bring about a factor of three reduction in the frequency of internally initiated accidents involving interfacing systems LOCAs, loss of offsite power, and failures of the auxiliary feed water system. The Staff estimated that the overall risk reduction attributable of these particular provisions was less than a factor of three due to the major contribution of external events, for which these changes make little or no difference. (Finding 2.0-6) Because the safety enhancement of the Director's Orders measures cannot be fully quantified does not mean that it is insignificant, however. One or more of the entire package of Director's Orders measures are applicable to most if not all of the accident sequences which were identified as contributing to the risk of the plant. (Finding 2.0-7)

The Necessity of Additional Safety Measures at Indian Point

Additional safety measures for the Indian Point units, other than those included within the Director's Orders, were subjected by the Commission to a screening test before they were entitled to be considered in this proceeding. The Commission's Order of September 18, 1981, 14 NRC 610, CLI-81-23, required that proposals for specific

safety measures (other than those referred to by the Director's Orders) should be accompanied by a preliminary showing that the measures being proposed would result in a significant reduction in that risk.

A number of additional safety measures proposals were advanced during the contention phase of the proceedings and, predictably, opinions varied widely as to whether the requisite threshold showings for admission of various proposed measures had been made. By our Order of November 15, 1982 we determined that contentions supporting three additional safety measures satisfied the Commission's screening test and were entitled to be considered in the proceeding. During the course of the hearings, further safety measures were evaluated in certain testimony offered by the Staff, and the Board asked that yet an additional set of safety measures be addressed by the parties.

With regard to the first part of the Commission's screening test for admission of contentions proposing additional safety measures, at the time of our contentions' rulings we rejected licensees' arguments that proponents were obliged to make some showing that the units posed significant risks in their present configurations. We rejected that argument and held that such a finding should abide the outcome of the proceeding. Now, however, after having heard all of the testimony on the subject of the risk to public health and safety posed by Indian Point, as

we have held in connection with Commission Question 1 above, the Board is unable to conclude that there exists a significant risk to public health and safety from operation of the plant. For this reason the Board would be entitled to abandon further consideration of additional safety measures for Indian Point, since we have found that one of the two preconditions established by the Commission has not been met. Nonetheless, we have before us an extensive record on proposals for several additional safety measures, and we are mindful that the Board has been directed to gather and present to the Commission all information relevant to the risk of Indian Point under various alternatives. Due to the basic investigatory nature of this proceeding, we believe that for us to evaluate such measures on their merits is consistent with the Commission's overall intent.

Filtered Vented Containment and Separate Containment Systems

Two devices were proposed by intervenors as means of mitigating the consequences associated with an overpressurization failure of the Indian Point containments following a core melt sequence. These devices, suggested in Contentions 2.1(a) and 2.1(d) respectively, are a filtered vented containment system (FVCS) and a separate

containment system (SCS). Since both devices are intended to reduce accident consequences from a single cause, we will consider them simultaneously.

A FVCS is a device which would be incorporated into the design of a containment building in order to permit controlled venting of a post-accident containment atmosphere. Filtered venting is a process which would entail the deliberate release of a portion of the containment atmosphere to the environment in a controlled manner through a system of filters. Such a pressure relief system would be actuated to reduce containment pressure which might under certain severe accident conditions otherwise lead to gross containment failure, and thereby allow an uncontrolled and unfiltered release of radionuclides into the atmosphere. (Finding 2.1-10) Installation of a SCS would seek to avoid the same containment failure addressed by a FVCS, but would instead attempt to maintain containment integrity by providing more containment volume and attendant heat sink and source term attenuation capability by means of interconnecting piping from the existing containment building to another large building. (Finding 2.1-11) As with a FVCS, this concept may have the potential to prevent or delay containment building failure and thereby reduce the consequences to the public from severe overpressurization accidents.

Testimony on the efficacy of FVCS and SCS was presented by sponsoring intervenors, licensees and Staff. An initial premise of witnesses favoring these devices was that mitigation devices such as these, which do not attempt to prevent or reduce the frequency of core melt accidents but rather seek to reduce their consequences, offer more promise in reducing risk than do preventive features because prospects for further reducing the probability of core melt accidents are limited. The argument assumes, of course, that further efforts to reduce risk are appropriate, a premise which we laid to rest in connection with Commission Question 1. However, the particular focus on core melt events themselves seems misplaced since there is no risk to the public from core melt accidents which are not accompanied by containment failure or bypass. There is accordingly no particular correlation between the magnitude of core melt frequency and the magnitude of risk. (Finding 2.1-14)

Since the FVCS and SCS would only serve to avoid or postpone the occurrence of containment overpressurization failures, the ability of the Indian Point containment structures to withstand post-accident pressures is quite obviously a crucial issue in evaluating the benefit which could be expected from such devices. The higher the containment failure pressure the fewer the number of accident sequences which could potentially cause a breach

of containment, thereby reducing the potential usefulness of a FVCS or SCS. (Finding 2.1-19) As noted in connection with Commission Question 1 above, we have determined that the record supports the conclusion that the Indian Point containment structures would withstand a pressure of 141 psia before onset of yielding, and thus prior to failure by breach or rupture as well. (Finding 2.1-20)

The containment overpressurization events for which a FVCS or SCS should be evaluated can be placed in three categories: (1) early or rapid overpressurizations, which involve hydrogen burns; (2) moderate overpressurization events, comprising primary system blowdown and molten core quenching; and (3) slow overpressurization events, consisting of core-concrete interactions and long-term decay heat events. (Finding 2.1-47) The parties appear to be in agreement that a FVCS or SCS would not be effective in preventing containment failure from rapid overpressurization events, inasmuch as the rate of pressure buildup would be such that the size of the containment penetration required to withstand the increase would be so large as to be impracticable.* (Finding 2.1-48) FVCS and SCS could

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* An intervenor analysis that a FVCS might offer protection against some hydrogen burn events was premised upon generic NRC analyses which assumed containment failure pressure to be 100 psia, which is substantially below the failure pressure of the Indian Point containments. Pressure spikes from all credible hydrogen burns would be below 141 psia, hence the studies assuming failure at lower pressures are inapplicable to Indian Point. (Finding 2.1-49)

potentially be effective in preventing containment failures from (2) and (3), moderate and slow overpressurization events. With regard to the first of these, however, the magnitude of the pressure resulting from moderate overpressurization events is not likely to fail the Indian Point containments and there is thus no need to consider mitigation devices for such scenarios. (Finding 2.1-50)

All of the parties agreed that neither the FVCS nor SCS would have any significant effect upon events which bypassed containment, which would bypass a FVCS or SCS as well. Among the bypass events for which these devices would be ineffective are the interfacing systems LOCA, steam generator tube rupture, and a failure to isolate containment. (Finding 2.1-51) This leaves the slow overpressurization events as ones which may be susceptible to mitigation by a FVCS or SCS. Since the record supports some potential benefit for such devices, we will next consider their engineering feasibility and attendant risks.

The proponents of a FVCS and SCS offered no specific design features for such devices, although there was a suggestion that an Indian Point FVCS could be generally similar to one planned for the Barseback nuclear plant in Sweden. (Finding 2.1-27) With respect to a SCS, use of the containment of the retired Indian Point Unit 1 was evaluated by Staff. (Finding 2.1-34) Provision for both manual and automatic initiation of both the FVCS and SCS was proposed. (Finding 2.1-21)

There are no FVCS or SCS installations at any commercial nuclear power plants in the United States. Such devices that are presently in existence are used with test facilities, liquid metal, and heavy water reactors. (Findings 2.1-24 and 25) The commercial nuclear power plant FVCS which is closest to fruition is that of the Barseback plant, which is still undergoing design.* Barseback is a boiling water reactor with a different containment volume and failure pressure, and the technology which could be derived from this and other installations is of varying but generally limited relevance to FVCS or SCS requirements for light water pressurized water reactors with large dry containments such as Indian Point. (Finding 2.1-27) The preponderance of the testimony supports theconclusion that FVCS and SCS technology is immature and unproven for operating reactors such as Indian Point. Since there is no operating experience or plant-specific design evaluation and testing of a FVCS or SCS for an analogous installation, there is no assurance that such systems would operate successfully under the specific conditions of interest at Indian Point. (Findings 2.1-27 and 28)

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* Testimony was uncontroverted that the decision to install a FVCS at Barseback was not accompanied by a detailed, probabilistic analysis to determine the risk of the unit with and without a FVCS.

Before a FVCS or SCS can be adequately evaluated there must be an assessment of the attendant risks which would accompany installation of such a device. Attendant risks are new risks, not previously existing, which would result from the installation of a particular mitigation feature. (Finding 2.1-37) Several attendant risks to a FVCS or SCS were identified. Since either device would involve a breach in the containment boundary (a departure from past NRC practice), the failure of such a system could under certain circumstances cause a release of radionuclides that would not otherwise have been released, creating in effect another bypass of containment. Were this to occur a low-consequence event could be significantly exacerbated. (Finding 2.1-38) Failure of a FVCS or SCS could occur through valve failures, or the effects of severe external events (which were found to be the dominant contributors to risk at Indian Point). (Finding 2.1-39) Other possible failure modes for a FVCS and SCS identified in the testimony included negation of containment spray injection system functioning, which is actuated by overpressurization, and failure of the FVCS or SCS interconnect due to hydrogen ignition in the device itself. (Finding 2.1-40) The parties also agreed that installation of a FVCS or SCS would increase the likelihood of basemat penetration. (Finding 2.1-52)

With respect to the possible use of the Indian Point Unit 1 containment as a SCS, the Staff did not recommend considering it further due to the significantly lower seismic capacity of the Unit 1 containment, and because of the potential need to install major new cooling and hydrogen control systems. (Findings 2.1-34 through 36) While many of the uncertainties about the effectiveness of such devices and their attendant risks could be accommodated or at least minimized by an adequate research and design program accompanied by a plant-specific PRA, such an undertaking would probably be long and costly. (Finding 2.1-45)

There was substantial disagreement among the parties as to the potential for risk reduction as a result of the installation of either a FVCS or SCS. Intervenors predicted large potential reductions in health effects and property damage from the installation of such devices, however these analyses assumed significantly lower containment pressure capability and correspondingly increased potential for early containment failure, and did not relate to the dominant contributors to risk (such as the interfacing systems LOCA which is unaffected by a FVCS or SCS) which were identified on a plant-specific basis for the Indian Point units in the IPPSS and Staff analyses. (Finding 2.1-53) Intervenor witnesses acknowledged a much lower risk reduction potential when the devices are modeled

with higher containment pressure capability approaching that determined for Indian Point. (Finding 2.1-55) The Board accordingly finds the potential risk reductions suggested by intervenors to be of significantly diminished value.

Staff testimony did present calculations of a potential factor of two and four reduction in early fatalities from Indian Point Units 2 and 3, respectively, through implementation of a FVCS. (Finding 2.1-63) As noted earlier, however, the Staff found that early fatalities could result from a slow overpressurization event only when very pessimistic emergency response assumptions were made. (Id.) With respect to latent health effects and property damage, Staff calculated approximately a factor of five reduction as a result of a FVCS or SCS. (Finding 2.1-64) Licensees' witnesses agreed that either device could potentially have a positive impact on latent effects and property damage from slow overpressurization events, but that their risk reduction value could not be reliably established absent a risk analysis of the proposed features joined to and interacting with the Indian Point plant and containment structures, and completion of presently ongoing source term research. (Finding 2.1-68) Both the licensees and the Staff agreed that when more realistic source terms are incorporated in the analysis, the risk reduction potential of both a FVCS

and SCS is correspondingly reduced, and the incentive for additional safety measures to lower risk diminished. (Finding 2.1-70)

The Staff's FVCS and SCS evaluation set forth only the potential risk reduction which could ideally be attributed to these devices, since no effort was made to assess the negative impact of the attendant risks of a FVCS or SCS on Indian Point plant safety. (Finding 2.1-65) Thus Staff's evaluation assumed the devices would be qualified to withstand the external events which are the dominant contributors to risk at Indian Point. (Id.) The record indicates that the attendant risks are not insubstantial or implausible, and that some of the scenarios which would accompany installation of these devices could result in a substantial increase in risk. (Finding 2.1-46) We are therefore persuaded that a careful probabilistic assessment of the risks attendant to a FVCS or SCS is essential to gain a true picture as to whether they would bring about a net reduction in risk.

While uncertainty surrounds the risk reduction potential of a FVCS or SCS, so too the true cost of such devices can only be accurately estimated after completion of an extensive site-specific and plant-specific design analysis. (Finding 2.1-72) The parties did, however, present very rough cost estimates, which were generally based upon the available literature. NUREG/CR-1410

estimated the cost of a FVCS at approximately $12 to $32 million, which does not include further research and development costs, qualification for external events, replacement energy costs for unit down time, or licensing delays. (Finding 2.1-73) The sponsoring intervenor estimated that as an upper bound figure, the cost of either a FVCS or SCS would not exceed $100 million, not including further necessary research and development or operating and maintenance costs. (Finding 2.1-74) Since the Unit 1 containment would require seismic upgrading, the preferred SCS alternative would be construction of a new containment structure, the cost of which would at least equal and most likely exceed that of a FVCS. (Finding 2.1-75)

Conclusion

Even if we were to disregard our conclusion under Commission Question 1 and assume that significant risks to public health and safety exist which would warrant taking additional safety measures, the record before us is not convincing that either a FVCS or SCS would be an appropriate choice. Since no such devices have been installed at any facility, knowledge regarding their design and functioning is not at the point where considered judgments can be made either about their risk reduction potential or their attendant risks. The Board is also persuaded based on the evidence that such devices would not

be cost effective in reducing risk at these units. The Staff did not recommend pursuing these mitigation techniques at Indian Point, and even incomplete quantitative assessments of their risk reduction potential fall significantly below the factor of 10 which has been proposed by the Staff as a threshold for considering a mitigation device to be of significant value. (NUREG-0850 at 4-2) Accordingly, the Board finds that the evidence on FVCS and SCS does not support a recommendation that either device be installed at the Indian Point facility.

The Use of Brackish Water Within Containment

Both of the Indian Point units employ brackish Hudson River water as the coolant in their fan cooler units within containment. It is alleged by Contention 2.2(a) that the operation of the fan cooler units should be changed so that brackish water would no longer be used. The purpose of the proposed change would be to combat alleged safety-related corrosion problems said to arise with the present configuration.

Each of the Indian Point units has five fan cooler units. Each unit consists of a main heat exchanger primarily made of tubing which cools the containment air, a fan to draw the air through the heat exchangers and circulate it throughout the containment, a demister to remove condensed moisture droplets from the air, and

various filters to clean the air. (Finding 2.2-7) The Unit 2 fan cooler system is supplied with cooling water through two 24-inch diameter cement-lined carbon steel pipes. Each fan cooler unit is then serviced by a separate 10-inch cement-lined supply and discharge pipe. (Finding 2.2-8) The fan cooler U-tubes, tube sheets, water boxes and inlet piping are all made from a copper/nickel alloy at Unit 2, and from iron-nickel-chromium-molybdenum alloys at Unit 3. (Finding 2.2-9) The fan cooler system removes heat loss from equipment and piping in the containment during normal plant operation, and in the event of an accident would remove sufficient heat from containment to prevent overpressurization. (Finding 1.1-150)

Prior to 1980, certain corrosive mechanisms and resultant leakage were identified in the fan cooler systems of the Indian Point units. At Unit 2, three types of corrosion occurred, including corrosion at imperfectly brazed fan cooler coil joints, pitting on the inside surface of the fan cooler tubing resulting from deposition of silt, and corrosion at welded joints of the cement-lined pipes which supply the fan coolers due to spalling of the lining at the welds. (Findings 2.2-12 through 14)

In 1980-81, significant modifications were made to the fan cooler systems at both Indian Point units, including replacement of all fan cooler coils, header and branch piping. (Finding 2.2-16) Both licensees submitted

testimony that corrosion problems associated with the use of brackish Hudson River water have been overcome. Con Edison, the licensee of Unit 2, also offered testimony that the particular types of corrosion which occurred in the fan coolers prior to their replacement were not associated with either the use of brackish water or the types of materials used in the fan cooler system. (Finding 2.2-15) Hudson River water is of sufficiently low salinity that it might well be classified as fresh water for most of the year, and other characteristics of Hudson River water at Indian Point, such as low pH level, low biological oxygen demand, and dissolved oxygen approaching saturation are indicative of water in which the alloys used in the two Indian Point units exhibit very low corrosion rates. (Finding 2.2-9) Nor is there anything unusual about the use of brackish or saline water for the cooling of steam electric power plants. Fifteen percent of domestic power plants and 33% of domestic nuclear power plants are located on brackish or saline bodies of water which are routinely used for cooling purposes. (Finding 2.2-6)

The sponsoring intervenor testified that the corrosive qualities of brackish water were such that it was necessary to eliminate it from the fan cooler system inside containment. The cooling system preferred by the intervenor is a closed cycle system using purified water within containment, which would be connected by a heat

exchanger with an external cooling source. (Finding 2.2-5) Although maintaining that such a system was necessary to eliminate corrosion, intervenor's witness had performed no feasibility study relating to implementation of a closed cycle system at Indian Point. (Id.) The Unit 2 licensee had developed a detailed design for a hybrid system similar to that proposed by the intervenor, but concluded that it, and indeed any system which avoided Hudson River water as the fan cooler system coolant, presented significant complexities in comparison with the present system. (Finding 2.2-4)

The starting point for the Board's analysis here, as with all additional safety measures proposed under Commission Question 2, is whether the particular measure under consideration offers promise for the "significant reduction in risk" contemplated by Question 2. However desirable a change in equipment or operation may be from some other standpoint, the Commission has directed that the particular focus of our inquiry here must be toward measures which might significantly improve levels of safety at Indian Point. The sponsoring intervenor was able to identify only two risks associated with the use of brackish water in the fan cooler systems: the wetting of the reactor vessel with cold water as a result of undetected leakage inside containment, and the wetting and subsequent shorting out of electrical circuits as a result of such

leakage. (Finding 2.2-3) However, intervenor's witness testified that he had performed no analysis of the risk posed by vessel wetting, and that he was not aware of any study of the effects upon risk of the wetting of electrical equipment such as that located within the Indian Point containments. (Id.)

Both the Staff and licensees testified that the brackish water issue is primarily a maintenance concern, with minimal implications to public risk, even in the event that fan cooler leakage were to go undetected for a long period of time. (Finding 2.2-24) Failures of the fan cooler system were included in both IPPSS and the Staff's risk analyses, and were found to have very little effect on risk. (Finding 2.2-25) Prior instances of fan cooler leakage were determined not to be significant from a public health and safety standpoint. (Finding 2.2-24) Staff testified that leakage from the containment fan coolers does not warrant requiring installation of a closed cycle system, and that such a modification would be warranted only if safe shutdown would be precluded in the event of a corrosion-related failure of a fan cooler unit, which was not the case. (Finding 2.2-26)

Conclusion

Based upon all of the testimony offered by the parties regarding this contention, the Board finds that fan

cooler system corrosion which occurred at the Indian Point units prior to substantial system modifications made in 1980-81 was not caused by the use of brackish water as the fan cooler system coolant, and that prior leakage did not result in any increase in risk to the public. System modifications which have been made by licensees provide reasonable assurance that fan cooler system leakage will be minimized in the future. In any case, since leaks -- even if substantial -- would not present any increased risk to the public, it follows that there can be no risk to the public from the continued use of the current system. The sponsoring intervenor for this contention was unable to establish any link between problems which it contended could be traced to the use of brackish water and an increase in the frequency of serious accidents resulting in the offsite release of radiation. The Board therefore finds that no significant reduction in the risk posed by the Indian Point plant would be achieved by the elimination of the use of brackish water in the fan cooler systems as proposed by Contention 2.2(a).

Steam Generator Tube Rupture Prophylaxis

Concerns about pressurized water reactor steam generator reliability have been prominent for a number of years. However, primarily as a result of a tube rupture event at the Ginna plant in January, 1982 the focus of

interest has broadened to include safety implications as well. On July 29, 1982 there was a meeting between members of the Staff and utility and vendor representatives of the Steam Generator Owners Group (SGOG) at which a number of proposed generic requirements intended to prevent steam generator tube ruptures were discussed. By Board Question 2.2.1, we asked whether the proposed requirements which were discussed at that meeting should be imposed upon the Indian Point licensees in order to reduce the risk of a steam generator tube rupture at that plant.

The Staff's proposed generic program sets forth a number of measures to preserve steam generator tube integrity, including prevention and detection of loose parts, stabilization and monitoring of degraded tubes, tube inservice inspection program, eddy current inspection techniques, maintaining primary to secondary leakage limits, a secondary water chemistry program, a condenser inservice inspection program, and upper inspection ports. The program also covers plant systems response and a standard technical specification limit for coolant iodine activity.

Each of the proposed SGOG generic requirements was examined in testimony submitted by the licensees and contrasted with programs already in place at each of the Indian Point units. A point-by-point comparison in the testimony of these programs with the proposed SGOG

requirements is persuasive that licensees' present programs are generally equivalent to, and as a whole exceed, the proposed requirements. (Finding 2.2.1-13) For this reason both the Staff and the licensees agree that the requirements proposed at the July 29, 1982 SGOG meeting should not be required for the Indian Point plants at this time. (Finding 2.2.1-15)

The Board's sua sponte interest in steam generator tube rupture prophylaxis is derived from the fact that steam generator tube failure would introduce primary coolant into the secondary side of the steam generators, thereby increasing the risk of release of radioactive material to the environment. Also, steam generator tube ruptures could compromise the availability of cooling water to the reactor core. (Finding 2.2.1-3) The Staff performed a generic analysis of steam generator tube failures leading to core melt and concluded that the total core melt probability from all causes involving steam generator tube failures is expected to be once in 250,000 reactor years. (Finding 2.2.1-4) This generic analysis was examined by the Staff and found to be applicable to the Indian Point units. Employing very conservative assumptions in modeling a steam generator tube rupture-initiated core melt, Staff estimated that such scenarios contribute approximately one percent of the aggregate offsite radiological risk. (Finding 2.2.1-5)

Other steam generator tube failures were examined in testimony as well. Westinghouse, the manufacturer of the Indian Point steam generators, analyzed a postulated event consisting of a simultaneous steam generator tube rupture and a large break LOCA. The results applicable to the Indian Point units showed only a very small increase in the peak fuel cladding temperature, showing that a large number of tubes would have to rupture in order to approach the limit established in 10 CFR 50, Appendix K. (Finding 2.2.1-6) The scenario of a steam generator tube rupture coincident with a stuck open valve was also analyzed in licensees' testimony and determined not to be a significant contributor to risk. (Finding 2.2.1-8)

Both the licensees and Staff offered testimony, which was not controverted, that implementation of the proposed SGOG requirements would not perceptibly change the risk of the Indian Point units as set forth in their respective analyses. (Finding 2.2.1-7) Since the risk attributable to steam generator tube ruptures is so small a contributor to the overall core melt risk at the Indian Point units, the adoption of the proposed SGOG requirements, even if they afforded some additional protection against tube ruptures not offered by the programs presently in place, could at best offer only a very small reduction in risk. (Id.)

Conclusion

The Board finds that steam generator tube rupture events comprise a negligible contribution to the risk posed by the Indian Point plant. Both licensees have had programs in place for some time which are similar to and as a whole exceed the proposed SGOG generic requirements. Since adoption of these proposed requirements would not enhance safety, there is accordingly no basis for imposing them upon the Indian Point licensees at this time.

Alternative Additional Safety Measures

At various times during the course of the hearings the Staff presented evaluations of possible additional safety measures beyond those advanced in the contentions and board question discussed above. These measures were generally ones which were intended to mitigate, rather than prevent, serious accidents. The Board did not receive testimony on these measures from any other party.

One measure evaluated by Staff was actually a triad of mitigation features consisting of glow plug igniters located within containment for the deliberate ignition of combustible gases, a passive containment building heat removal system to control overpressurization consisting of heat pipes, and a system to insure the flooding of the reactor cavity to prevent basemat

penetration. (Finding 2.1-78) Among the release categories that dominate risk at Indian Point, only the slow co..ainment overpressurization events would be mitigated or eliminated by this triad of features. (Finding 2.1-82)

The intended purpose of glow plug igniters would be to mitigate the hydrogen burn containment failure mode under post-accident conditions by providing a means of controlled burning of combustible gases so as to render the containment atmosphere inert. (Finding 2.1-79) This would reduce the likelihood that thermal or pressure loadings might cause failure of vital equipment or containment integrity. (Id.) Two of the attendant risks of such devices identified by Staff are the possibility of an increase in containment pressure, and equipment failures due to localized burning. (Id.)

The second element of the Staff triad was a heat pipe system for the passive removal of heat from containment, also for the purpose of controlling post-accident environment containment overpressurization. A heat pipe is a closed pipe system which contains a working fluid. Vaporization of the fluid within containment and subsequent condensation outside containment releases heat from within containment to the atmosphere. (Finding 2.1-80) No heat pipe system has ever undergone a prototype or design demonstration program for any nuclear

power plant. (Id.) The heat pipe wall thickness evaluated by Staff was three millimeters, necessarily thin so as to provide for effective heat transfer. The resultant fragility of the heat pipes is such that there would be significant engineering challenges to developing a practical application of the heat pipe concept to the Indian Point containments and qualifying it to withstand internal and external events. (Id.) Physical breaks or ruptures to the heat pipes caused by the external events found to dominate risk at Indian Point would provide an exit pathway from containment which could exacerbate accident consequences rather than mitigate them. The design objective of the heat pipes was to reduce containment pressure to a level below design pressure within 12 hours of accident initiation. The number of pipes which would be necessary to accomplish this is such that it might be necessary to cut through layers of containment reinforcing steel to create the number of penetrations necessary to accommodate the heat pipes. (Id.)

The last element of the Staff triad was a system which would insure flooding of the reactor cavity under post-accident conditions. (Finding 2.1-81) Such flooding would establish a coolable debris bed in the reactor cavity, thereby limiting core/concrete interactions and reducing the likelihood of basemat penetration. (Id.) As

with the other features, the flooding measure has an attendant risk, in this instance reduction of effective containment volume and a consequent increase in the probability of overpressurization failure. The Staff's testimony found that a flooded cavity would be counterproductive if it were not installed in conjunction with the second element of the triad, heat pipes. (Id.)

Staff estimated that the cost of the triad features would likely be several tens of millions of dollars. (Finding 2.1-78) After an attempt to account for the negative characteristics and failure rate of each element of the triad, the Staff determined that the risk reduction from the triad in terms of latent health effects was only a factor of five. (Finding 2.1-82) As noted above, this is below the factor of ten threshold for consideration of risk reduction backfits recommended in earlier Staff studies.

Staff estimated the cost-effectiveness of the triad for the Indian Point plant using four different mathematical algorithms which were intended to estimate the dollar value of risk reduction. Based upon this and its engineering evaluation, the Staff did not recommend implementation of the triad at Indian Point. (Finding 2.1-83) In fact, in its testimony before the Board the Staff recommended generally that the Commission not issue orders aimed at enhancing the ability to mitigate severe

reactor accidents at Indian Point. (Finding 2.1-85) The Staff acknowledged that this represented a departure from its 1980 Task Action Plan for Indian Point and other populous sites, which had placed heavy emphasis on hydrogen control, filtered venting and core retention devices.

As explained by the Staff, it was now clear that the plants as built provided effective mitigation of the offsite consequences of core melt accidents, and that slow overpressurization failures of containment would not occur with an operating containment heat removal system such as the presently existing fan cooler and spray systems. (Finding 2.1-84) The basis for the Staff's position in the hearings was therefore that most of the desirable attributes of the mitigation devices considered earlier were already present in the Indian Point plant designs, that the risk of the Indian Point plant is small and within the range of risks posed by other plants licensed to operate by the Commission, and that other ongoing generic inquiries applicable to all plants provided a more appropriate vehicle for pursuing design improvements. (Findings 2.1-86 and 92)

The Staff did, however, recommend that the licensees be ordered to implement a Safety Assurance Program (SAP), the objective of which would be to implement and maintain PRA recommendations for the conduct of plant operations such that the level of risk of the Indian Point

plant would remain consistent with that assessed by Staff. (Finding 2.1-87) Staff acknowledged that an updating of IPPSS and the translation of the lessons learned from such an update into the conduct of operations would accomplish most of the objectives of the SAP. Program startup costs were estimated by Staff to be a few million dollars, with continuing costs of a few hundred thousand dollars per year. (Finding 2.1-89) Staff also suggested that although the residual risk is not large, such a program would likely be cost-effective. (Id.) In arriving at this conclusion, however, no monetization benefit calculation was performed for the SAP similar to that performed by the Staff for other additional safety measures. (Finding 2.1-90) Staff did not perform an intensive evaluation of the effectiveness of such a program, nor did it review whether existing licensee activities would already satisfactorily address the intended purposes of the SAP. (Id.) Staff also admitted that the incentive for and likely benefit of the program would be diminished if lower source terms were used for modeling severe accidents. (Finding 2.1-91)

Conclusion

Based on the record before us, the Board agrees that the triad of mitigation features evaluated by the Staff are mitigation concepts which should not be required for the Indian Point plant. The record on this point, which derives entirely from the Staff, clearly indicates that the triad was evaluated only in a very preliminary way, and was not pursued after attendant risks and further major design hurdles loomed large. Since the risk of Indian Point is low and within the range of risks posed by other plants, accident mitigation and risk reduction are most appropriately considered in the context of generic programs for all operating reactors.

Similar logic requires us to avoid recommending that the licensees be required to implement the Safety Assurance Program concept. The licensees have shown commendable initiative in recently incorporating several plant modifications shown by the IPPSS to achieve significant reductions in risk, and there is no reason to believe that they would not continue to do so. No good reason has been advanced for singling out the Indian Point plant for such a program, and in fact the conclusion that risk at that site is within the range of risks posed by other plants would appear to preclude such a justification. Single site implementation of a major

program such as that discussed by Staff may also conflict with the Commission's current views on backfitting. The Commission's current backfitting regulation, 10 CFR § 50.109, requires a showing of "substantial, additional protection which is required for the public health and safety" before the addition of a new "system"* may be ordered at an operating plant.

Commission Question 2 which we are charged with following requires that there be both a significant risk to the public without the additional measure being considered, and also that implementation of that measure would result in a significant reduction in that risk. We have already concluded, based in part upon analysis supplied by the Staff, that the Indian Point units do not pose a significant risk in their present form. As to the reduction in risk which would be afforded by the SAP, the Staff's evaluation was wholly insufficient for us to find that it would be significant. We accordingly find that the evidence before us on both the triad of mitigation features and the SAP does not support a recommendation that they be adopted at Indian Point.

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* Whether a safety assurance or similar program would be considered a system for purposes of 10 CFR § 50.109 as presently drafted has not been previously addressed. The Board is aware that considerable attention is currently being given within NRC to revising the backfitting regulation, and that the recent Commission guidance memorandum to the Staff (COMSECY -83-3) dated June 22, 1983 defines the type of backfitting requiring either licensee consent on an elaborate appeal process as any staff-proposed requirement that involves a new staff position or change in an existing position with respect to the licensee."

CONFORMANCE WITH NRC/FEMA EMERGENCY PLANNING GUIDELINES IN THE VICINITY OF INDIAN POINT

Although the Commission indicated that risk issues, and particularly comparative risk, was its "primary concern" in the conduct of this proceeding, it did charge the Board with assembling a record on certain emergency planning issues as well. Commission Question 3 asked the parties to address the current status and degree of conformance with NRC and FEMA radiological emergency planning guidelines in the vicinity of Indian Point. Even in this area the Board was asked to relate its inquiry to risk issues, and in particular we were asked to attempt to establish what the minimum number of hours warning would be for an effective evacuation of the plume exposure pathway emergency planning zone.

The topic of offsite emergency planning in the area surrounding the Indian Point plant was the primary focus of testimony offered by intervenor groups and interested states. During the hearings the Board by its rulings attempted to distinguish between instances where compliance with guidelines was being addressed from those where skepticism about the sufficiency of the guidelines themselves was being raised. While we may not always have been successful in making this clear-cut distinction during the course of the hearing we are constrained to do so in our decision by the directives of the Commission.

Furthermore, our findings in the area of radiological emergency planning must also be developed and interpreted in light of the Commission's stated overall objective for this proceeding, which is to determine the extent to which the population in the vicinity of Indian Point affects the risk posed by the plant as compared to the range of risks posed by other operating nuclear power plants. The Board has not viewed itself as supplanting in any way the proposed 44 CFR 350 radiological emergency plan accreditation process, the routine periodic exercises to evaluate emergency response capabilities at Indian Point, or the post-exercise appraisals and so-called 120-day clock enforcement mechanisms of the Commission's regulations. Indeed, all of these activities have been ongoing during the course of our proceedings and will doubtless continue long after it is over. While these activities have provided valuable input into this proceeding, ours has been an independent and collateral investigation of primarily risk issues in which certain aspects of emergency response have played a significant role.

Overview of Radiological Emergency Planning Activities in the Vicinity of Indian Point

The Commission first posed its Question 3 regarding the status of radiological emergency planning in the area around Indian Point in its January 8, 1981 Order. (13 NRC 1) This was just a few months after the Commission's greatly expanded radiological emergency planning regulations had gone into effect, and naturally the status of emergency planning was quite different from what it was at the close of our record on April 29, 1983. The record of this proceeding clearly establishes that emergency planning is a dynamic and ongoing process. Because we are constrained to address the "current status" of this process,* we believe that a background description

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* We interpret the Commission's directive to report on the "current status" of Indian Point emergency planning to mean on or about April 29, 1983, the last day of our evidentiary hearings. This has affected the value of a good deal of the emergency planning testimony, most of which was filed in June 1982. Opportunities extended by the Board to file supplemental or updated testimony, such as in our February 7, 1983 order, were not taken in many instances. There have also been Indian Point emergency planning developments occurring subsequent to the close of our hearings which we are unable to address given our evidentiary format. As we noted in our November 15, 1982 order at pp. 17-18, the Commission can consider other information which it obtains on Indian Point in addition to the Board's decision and recommendations.

of state, licensee and county efforts to achieve compliance with the Commission's revised emergency planning regulations will be helpful to an understanding of the present picture.

When the NRC proposed amendments to its emergency planning regulations, 10 CFR Part 50 and App. E, in November 1979, Indian Point was one of the few operating nuclear power plant sites to have comprehensive emergency planning in place. Prior to the Three Mile Island incident, Con Edison had already implemented emergency planning, which included a plan coordinated with the State and Westchester County, an on-site meteorology system, a backup and primary emergency control center, field monitoring teams with radio communication equipment, and agreements with various organizations to provide assistance in the event of an incident. Con Edison also conducted ongoing emergency planning training and drills with local fire, ambulance and police. (Finding 3.1-238)

In December 1979, Con Edison and the New York Power Authority retained a large civil engineering firm to prepare evacuation time estimates. In early 1980 the licensees proposed to the four counties around Indian Point that licensees fund the activities of consultants to develop county and state radiological emergency response plans which would conform to the NRC's new regulations. Consultants were retained with the assistance of emergency

planners from the state and the counties. Although compensated by the utilities, the consultants acted as consultants for the state and the four Indian Point counties.

The record shows that the consultants actively solicited the participation and cooperation of State and local government officials. (Finding 3.3-6) Throughout 1980 and much of 1981, the consultants preparing the plans and State representatives met frequently with officials of the four Indian Point counties. These county officials included those responsible for civil defense, law enforcement, health, schools, public works and highways, fire and rescue, ambulance corps, social services and transportation. Planners also held frequent meetings with officials from schools, nursing homes, hospitals and various public agencies in the four counties to assess evacuation needs and to inventory equipment and personnel that would be available to aid in responding to an accident. All draft plan materials were also discussed with the New York State Radiological Emergency Preparedness Group (REPG) and the Office of Disaster Preparedness. The REPG is a specialized radiological emergency preparedness organization with a permanent staff which functions as an arm of the State's Disaster Preparedness Commission, and has lead responsibility for all offsite radiological emergency planning. (Finding 3.1-11)

Radiological emergency response plans for the State of New York and the four counties surrounding Indian Point were submitted by the State to NRC on December 30, 1980, and revised county plans were submitted on August 17, 1981. (Id). New York State Exhibit 3 represents the offsite radiological emergency response plans (RERPs) for Indian Point as they existed in August 1981. (Id.) This nine-volume Exhibit includes the one-volume state plan (i.e., both the generic RERP as well as the site-specific portion of the state plan), as well as two-volume RERPs for each of the four counties. (Davidoff, Tr. 11295-11298) These plans were prepared under the direction of the State's REPG and the New York State Disaster Preparedness Commission pursuant to that body's responsibilities under Article 2-B of the New York State Executive Law. (Finding 3.1-11) Mr. Davidoff testified that New York State Exhibit 3 was developed in accordance with the guidelines contained in NUREG-0654. (Id.)

In April 1981, NRC transmitted to the licensees a deficiency notice which expressed concern over whether in the event of a radiological emergency State officials and the Governor would have sufficient authority to insure the participation and direct the activities of county and local workers. State planners also wished to initiate a permanent program to provide continuous funding for radiological emergency preparedness to replace the

contractual arrangements started in April 1980. On July 9, 1981 the New York State Legislature passed Chapter 708 of the Laws of 1981.* This legislation gives the Governor or his designee complete powers to direct the participation and activities of county officials and other local workers in the event of a radiological emergency.

This legislation also imposes an annual fee of $250,000 per reactor upon utility licensees in New York State to be used for radiological emergency preparedness purposes at both the state and local levels. In the approximately two years this law has been in effect, the State Disaster Preparedness Commission and the REPC have received fees amounting to more than $1,000,000 from the Indian Point licensees alone. (Finding 3.1-84) From the funds received through utility assessments, the State REPC makes annual direct cash grants for radiological emergency planning to affected counties throughout the State. (Finding 3.1-84) In addition to these cash grants, the State has to date also purchased and provided to the four Indian Point counties from utility payments under the funding legislation over $73,000 in equipment, including 950 dosimeters and 5500 TLDs, as well as providing training and assistance of State personnel. (Findings 3.1-119 and 3.1-144)

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* Now codified as Executive Law, Art. 2-B, McKinney's Consolidated Laws of New York.

In 1980, the utilities began to receive equipment requests directly from the four counties. Two complete mobile radiation monitoring kits were delivered to each county in 1980, and with the assistance of county officials, including Rockland County, a network of fixed-site Ludlum and Reuter-Stokes radiation monitors was installed in each county. The utilities also agreed to supply extensive communications and computer equipment to each of the four counties, and by mid-1981, the Emergency Operations Center in each county was supplied with a computer to access meteorological and radiation data from Indian Point. (Finding 3.1-83) A dedicated Hotline and a backup system were also installed at utility expense which permit instantaneous communications among the State, four counties and the site. (Id.) In December 1982, Con Edison and the Power Authority made a further voluntary payment to the State, beyond that required by law, of $210,000, to be used in the four Indian Point counties. (Finding 3.1-85)

Starting in March 1982, the State REPG has provided radiological emergency response training regularly to state and local workers in the four counties surrounding Indian Point. To date, State-sponsored training has been given to a total of 4,892 attendees. (Finding 3.1-144) Those who have received training include police, firemen, ambulance and rescue teams, radiological exposure and decontamination workers, bus drivers, hospital workers, and school superintendents.

On March 3, 1982 the initial full-scale emergency planning exercise was conducted at Indian Point, and on May 27, 1982 FEMA issued an assessment of the offsite portions of the exercise. (Finding 3.1-13) This document, entitled "Post Exercise Assessment, Exercise of the New York State and Westchester, Putnam, Rockland and Orange Counties Emergency Plans for Indian Point Nuclear Cenerating Station" (1982 Post Exercise Assessment), is Attachment C to FEMA's Initial Testimony. (FEMA Testimony, p. 5) The 1982 Post Exercise Assessment FEMA asserted that at the time of the March 3, 1982 exercise there were significant planning deficiencies in five of the fifteen planning standards applicable to offsite emergency planning. (Id.) The State's REPC submitted an item-by-item plan for remedial action in response to FEMA's 1982 Post Exercise Assessment. (Finding 3.1-14)

On July 30, 1982, at the request of the NRC, FEMA issued another report entitled "Interim Findings on the Adequacy of Radiological Emergency Response Preparation of State and Local Governments of the Indian Point Nuclear Power Station" (Interim Findings). The Interim Findings informed the NRC of the significant deficiencies in the five planning standards noted in the May 27, 1982 Post Exercise Assessment. (Finding 3.1-15) As a result of the Interim Findings, on August 3, 1982 the NRC issued an order under the emergency planning enforcement procedures, 10 CFR

§ 50.54(s)(2)(ii), which notified licensees that unless the significant deficiencies were corrected within 120 days, the NRC would consider whether enforcement action was appropriate. (Id.) Among the deficiencies cited was the non-participation in the emergency planning process of Rockland County, one of the four counties in the vicinity of Indian Point.

In response to this order a working group was established at the request of NRC and FEMA, which consisted of New York State, the licensees and representatives of the four counties (Rockland County's participation in this work was limited to being an observer). This working group met regularly with NRC and FEMA in an attempt to resolve the deficiencies noted in the Interim Findings. (Id.) The final report of this group, New York State Exhibit 7, is entitled "Interim Findings, Indian Point Radiological Emergency Preparedness Remedial Action Schedule." This document presents the State of New York's response to each of the 34 elements of the five significant deficiencies which had been claimed by FEMA. (Id.)

On December 16, 1982 FEMA issued its "Update Report on the Status of Remedial Actions Cited in the July 30, 1982 Interim Findings on the Adequacy of Radiological Emergency Response Preparation of State and Local Governments at the Indian Point Nuclear Power Station." (December 16, 1982 Update Report) This document is

attached to FEMA's Supplemental Testimony, ff. Tr. 14741. The December 16, 1982 Update Report concluded that substantial progress had been made in upgrading radiological emergency planning in the vicinity of Indian Point, and that all of the 34 significant deficiencies noted in the Interim Findings had been adequately addressed with the exception of two areas. First, FEMA concluded that the absence of contractual arrangements with commercial bus drivers in Westchester County constituted a significant deficiency in NUREG-0654 Planning Standard J "Protective Actions" (i.e., 10 CFR § 50.47(10)). Second, FEMA concluded that due to Rockland County's lack of involvement in the emergency planning effort and the diavowal by that County of the emergency plans that had been prepared for it, a significant deficiency existed with regard to NUREG-0654 Planning Standard P "Responsibility for Planning Effort" (i.e., 10 CFR § 50.47(b)(16). (Finding 3.1-16)

On March 9, 1983 a second full-scale exercise of emergency response plans for Indian Point was conducted. On April 14, 1983, FEMA issued its 1983 Post Exercise Assessment, which noted a general, marked improvement in the level of preparedness in offsite emergency planning at Indian Point over that displayed in the March 1982 exercise. (F3.1-17) Nevertheless, based upon his review of the 1983 Post Exercise Assessment, Mr. Petrone

determined that the two significant deficiencies noted in the December 16, 1982 Update Report, that is, the lack of contractual arrangements with Westchester County bus drivers and the question of Rockland County's involvement in emergency planning, remained. (Id.)

From the time of their submittal to FEMA in August 1981, the state and four county RERPs have undergone extensive revisions to reflect comments of FEMA and others as well as to incorporate changed circumstances. New York State Exhibit 6 constitutes revisions that were made to the State's Radiological Emergency Response Plan which have been submitted by the State to FEMA. (Finding 3.1-18)

Following its decision to reject the original Rockland County Radiological Emergency Response Plan the County set about to develop a revised plan. The New York State Radiological Preparedness Group provided assistance in this effort. The County has produced a draft plan which the State has adopted as a compensating measure to be used until the County completes the development of its own plan. (Davidoff, Tr. 1136[illegible], [illegible]364) It was this plan which was exercised during the [illegible] 9, 1983 exercise. Mr. Davidoff testified that the adoption of the draft plan as compensating measure was done with the consent of the County. (Id.)

Assessing Indian Point Planning Against the Sixteen Planning Standards of 10 CFR § 50.47

The Commission's sixteen standards for assessing the adequacy of nuclear power plant radiological emergency planning are set forth in 10 CFR § 50.47(b) and Appendix E to 10 CFR Part 50. Further guidance is contained in a document entitled "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," NUREG-0654, FEMA-REP-1, ("NUREG-0654"). It is against these standards and NUREG-0654 guidance that we have been asked in connection with Commission Question 3 to evaluate the sufficiency of the emergency plans for Indian Point. Intervenors also advanced Contention 3.1, which alleges that the Indian Point plans fail to meet any of the sixteen planning standards in 10 CFR § 50.47(b).

The testimony presented in this proceeding by the State of New York through the Director of the State's REPC and also the testimony of FEMA both assess the level of compliance of the New York State and county RERPs for Indian Point with federal planning standards. The licensees' position was that the requirements of the planning standards had been satisfied, and intervenors, as proponents of Contention 3.1, generally alleged that the requirements were not met. However, neither licensees nor

any parties to this proceeding other than the State's REPG and FEMA attempted in their testimony to conduct a detailed assessment of Indian Point plan compliance with the planning standards. Generally speaking, the testimony of the intervenors was directed towards the other specific contentions which were advanced and only infrequently cited specific regulatory requirements. The great majority of witnesses sponsored by intervenors professed no familiarity with the NUREG-0654 guidance document or the applicable regulations. Witnesses responsible for emergency planning in two of the surrounding counties, Orange and Putnam, did testify that the plans for their counties comply with NUREG-0654 guidance. Although witnesses for Westchester and Rockland Counties with responsibilities similar to those of the witnesses from Orange and Putnam Counties did testify regarding the level of emergency preparedness in their respective counties, no attempt was made by these witnesses to measure the emergency plans for their counties against the standards of 10 CFR § 50.47(b).

This leaves us with primarily the State REPG testimony and the FEMA testimony as specifically addressing compliance with the planning standards. The State of New York's position is that the plans are in substantial compliance with all of the standards. (Finding 3.1-29) FEMA, by the time when we closed our record on April 29, 1983, was generally in agreement with the State REPG except

in two significant respects. As we noted above, FEMA contended the alleged lack of assurance of bus availability for evacuation in Westchester County during a radiological emergency, and Rockland County's non-participation in the emergency planning process on a coordinated basis with the state and other counties, constituted significant deficiencies regarding two of the sixteen 10 CFR § 50.47(b) planning standards.

In determining whether a particular standard is met this Board must make a reasoned, qualitative judgment. A standard can be met even though there are areas where improvements can and should be made. Similarly, as FEMA testified, it is possible for one or more of the particular NUREG-0654 criteria elements which are used as guidelines in assessing compliance with a standard to be not satisfied and still find that planning standard is satisfied.* (Finding 3.1-20) The evaluation criteria amplify and

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* The distinction between the guidance of NUREG-0654 and the planning standards of 10 CFR § 50.47(b) is a significant one. The former provides only suggested means of achieving regulatory compliance, and is not itself mandatory. See Gulf States Utilities Co. (River Bend Station Units 1 and 2), ALAB-444, 6 NRC 760 (1977). This is more than a matter of scholarly concern here since FEMA testified that its assessments of planning sufficiency at Indian Point following the 1982 and 1983 full-scale exercises were based upon NUREG-0654 criteria rather than the planning standards of the regulations. (Keller, Tr. 14860) In the Board's view, this distinction affects the weight which can be given to FEMA's testimony as we evaluate planning not against the guidance document but rather against the regulations themselves.

"expand on the standards" (FEMA Testimony, p. 3) set forth in 10 CFR § 50.47(b). A determination whether an emergency plan meets a particular overall planning standard requires an examination of the elements embodied within the particular planning standard. Because of the broad claims of Contention 3.1 that none of the planning standards are met at Indian Point, we will address them seriatim.

Planning Standard 1

Planning Standard 1 of 10 CFR § 50.47(b) requires an assignment of primary responsibility for emergency response among various response organizations, an establishment of respective responsibilities, and an identification of staff within each principal response organization. The onsite plans of each licensee and the State and four County RERPs do delineate the respective responsibilities of licensees, state and local governments, and the functions and staffs of each of the various organizations. (Finding 3.1-22) The plans also identify the various support organizations which would be called upon in an emergency including branches of the federal government and private organizations, such as the Red Cross. FEMA and the NRC found compliance with this standard to be deficient until the passage of the Chapter 708 legislation by the State as referred to earlier.

(Finding 3.1-24) Since that time, there have been no claims of significant deficiency under this planning standard, although the FEMA December 16, 1982 Update Report found that there were some deviations from NUREG-0654 evaluation criteria. (Finding 3.1-26) No probative evidence was offered which alleged non-compliance with the planning standard. FEMA's findings with regard to overall compliance with Planning Standard 1 are in agreement with those of the Director of the State's REPC, and also those of Orange and Putnam County officials as pertain to their county plans. (Finding 3.1-29) In light of this expert testimony and in view of the absence of testimony alleging noncompliance, the Board finds that this standard is met.

Planning Standard 2

Planning Standard 2 of 10 CFR § 50.47(b) requires the development of clearly defined responsibilities for onsite licensee personnel to provide initial accident response, that a capability to augment initial response is available, and that established means of interfacing between on and offsite activities are specified. Each of the licensees submitted detailed testimony in an attempt to establish the sufficiency of onsite emergency planning at each unit. This testimony was not disputed by any other party, and the licensees' witnesses who sponsored this

testimony were not asked any questions on cross-examination. With the exception of Contention 3.4 discussed below, we believe it is fair to say that no party challenged the sufficiency of licensees' onsite emergency planning under the applicable planning standards of 10 CFR § 50.47(b), and also Appendix E. Other than the testimony of licensees, the only other party to submit evidence on the level of compliance of licensees' plans with planning requirements was the Staff. Staff addressed each of the planning standards applicable to licensees' emergency plans and concluded that they were in compliance. The record reveals that subsequent to the adoption of the Commission's new emergency planning regulations the licensees have added radiation monitoring, emergency notification and communications equipment exceeding that which is required. (Finding 3.1-196) They have also adopted procedures for the notification of offsite authorities which exceed requirements. The presence of two licensee organizations at the site provides opportunities for backup and supplemental support in the event of an emergency at either unit. In our Findings we have carefully reviewed the licensees' emergency plans under each of the planning requirements. Based on that review the Board finds that the licensees' emergency plans are in compliance with the planning standards of 10 CFR § 50.47(b) as well as the standards of Appendix E to 10 CFR Part 50.

Planning Standard 3

Planning Standard 3 of 10 CFR § 50.47(b) deals with arrangements for requesting and accommodating State and local staff at the near-site Emergency Operations Facility, as well as identifying organizations capable of augmenting the response to an emergency. The evidence establishes that the EOF is an adequately equipped facility. (Finding 3.1-79) The State and county plans make provisions for calling upon Federal support under the Federal Radiological and Monitoring Assistance Program. FEMA's 1983 Post Exercise Assessment noted that EOF facilities and resources were good, and that active participation by all state, county and utility organizations at the EOF was outstanding. The Staff also testified that the licensees are in compliance with this planning standard. (Finding 3.1-184) No testimony was offered by intervenors or interested states which contended that the licensees were not in compliance with this standard. Based on the uncontroverted evidence, the Board concludes that the standard is met.

Planning Standard 4

Planning Standard 4 of 10 CFR § 50.47(b) requires licensees to implement a standard classification and action level scheme to indicate the seriousness or potential seriousness of accident or pre-accident conditions, and requires state and local plans to rely upon licensees' scheme in determining offsite response measures. The record indicates that both onsite and offsite plans incorporate the four emergency action levels specified in NUREG-0654, specifically notification of unusual event, alert, site area emergency, and general emergency. (Findings 3.1-40 and 3.1-186) The State and county plans provide for reliance on licensees' emergency action levels in connection with making protective response decisions. Each of the licensees have specific written procedures to be used in determining the protective actions to be recommended to offsite authorities. (Finding 3.1-186) Staff testified that the requirements of Planning Standard 4 and the related Appendix E provisions have been met by the licensees, and none of the FEMA documents or testimony suggest any deficiencies in compliance with this standard. (Findings 3.1-40 and 3.1-188) In light of the above and the absence of any evidence to the contrary, the Board finds that this planning standard has been met.

Planning Standard 5

Planning Standard 5 calls for the establishment of procedures by the licensee to notify emergency personnel and offsite response organizations, for the development of preplanned initial and followup messages to response organizations and to the public, and for the establishment of a capability to promptly notify and instruct the public within the plume exposure pathway Emergency Planning Zone. Both the onsite and offsite emergency plans contain elaborate notification procedures. (Findings 3.1-43 and 3.1-190) The State and county RERPs provide detailed procedures for notifying emergency response personnel upon receipt of notification by the licensee. There are numerous, redundant communications systems at Indian Point which permit prompt notification and communication to and among all affected state, local and federal authorities. (Finding 3.1-191) Initial notification of offsite authorities by the licensee would be accomplished through the Radiological Emergency Communications System (RECS), which is a direct, dedicated telephone system connecting licensees and governmental authorities. (Finding 3.1-43) There are independent radio communications links between the EOF, field personnel, corporate headquarters and governmental authorities as well. (Finding 3.1-58)

Messages have been prepared which relate the nature of the event, whether there has been a release, proposed protective actions, and meteorological conditions. (Finding 3.1-43) Each licensee has a page/party and a conventional telephone system to notify licensee emergency response personnel. (Finding 3.1-191)

A matter of considerable interest following the 1982 exercise at Indian Point was the siren system, a system purchased and installed by the licensees which is designed to provide early notification to the public. Difficulties were encountered with the siren system during the 1982 exercise, following which the licensees conducted an extensive program of system enhancement. Existing sirens were upgraded and in some instances relocated, and additional sirens were installed. In addition, tone alert radios were provided by licensees to schools, nursing homes and other special facilities. (Finding 3.1-44) During the 1983 exercise, all sirens were found to have operated properly, and both FEMA and the Staff modified previous statements of deficiency and concluded that the siren system now meets the planning standard. (Finding 3.1-47)

Based upon its review of the record, the Board concludes that Planning Standard 5 has been met.

Planning Standard 6

Planning Standard 6 of 10 CFR § 50.47(b) requires that provisions exist for prompt communications among principal response organizations to emergency personnel and the public. Staff testified without contradiction that the licensees' systems, comprising the RECS, radio and telephone systems described above and also a NAWAS (National Warning System) and NRC Emergency Notification System resulted in the satisfaction of this standard. (Finding 3.1-196) We concur in that judgment. Communications between offsite authorities can be conducted with the foregoing systems as well as with the Radio Amateur Civilian Emergency Services (RACES), an organization of "ham" radio operators who have been trained to respond to radiological and other emergencies. (Finding 3.1-58)

While the State and principal county emergency response facilities such as the Emergency Operation Centers (EOCs) are equipped with extensive communications equipment, a number of local law enforcement officials testified that not all police departments within a particular county are connected via a Multi-District Radio, and that this could impede communications in the event of an emergency. The record does reflect, however, that such officials have generally been able to respond effectively to past non-radiological emergencies. (Finding 3.1-62)

Additionally, the testimony indicates that there is now a dedicated hotline in use among municipal police departments which is unaffected by the differences in radio frequencies which hamper certain types of radio communications. (Id.)

In considering the sufficiency of existing governmental communications systems under Commission radiological emergency planning standards we confront a more general issue: whether our regulations were intended to require improvements to the infrastructure of local governments, or whether they were instead intended to take everyday governmental capabilities and facilities basically as they were found. While specialized radiological training and dedicated equipment such as dosimeters are one thing, the Commission's emergency planning regulations take on quite a different complexion if interpreted to require new bridges, new hospitals, or in this instance, new communications equipment.

The Commission has fortunately provided recent guidance in this difficult area of applying emergency planning requirements in such situations. In Southern California Edison Co. (San Onofre Nuclear Generating Station Units 2 and 3), CLI-83-10 (April 4, 1983), the Commission considered an assertion by intervenors that additional medical facilities not then in existance were necessary in order to achieve compliance with the emergency planning standards. The Commission rejected that interpretation of its regulations as follows:

> "It was never the intent of the regulation to require directly or indirectly that state and local governments adopt extraordinary measures, such as construction of additional hospitals or recruitment of substantial additional medical personnel, just to deal with nuclear plant accidents. The emphasis is on prudent risk reduction measures. The regulation does not require dedication of resources to handle every possible accident that can be imagined. The concept of the regulation is that there should be core planning with sufficient planning flexibility to develop a reasonable ad hoc response to those very serious low probability accidents which could affect the general public." CLI-83-10 at p. 8 (emphasis in original).

Based upon the foregoing Commission decision we must reject the claimed inadequacy of everyday law enforcement communications equipment as a grounds for deficiency.

Another issue which arose under both this planning standard and the prior one related to the adequacy of the Emergency Broadcast System (EBS) messages. One intervenor witness proposed that in the event of an emergency, different EBS messages be transmitted for different audiences based on psychological, social and other distinctions. The Board finds that substantial confusion could accompany such a complex approach to public communications under conditions of a possible accident. (Finding 3.1-54) Alternatively, we find that a multiple EBS message system is not required for compliance with our planning standards.

FEMA's 1983 Post Exercise Assessment reflects improvements in compliance with Planning Standard 6, and improvements since the 1982 exercise were noted. (Finding

3.1-60) FEMA's recommendations for further improvements in communications were quite limited. Based upon our review of the record the Board concludes that the requirements of this planning standard have been satisfied.

Planning Standard 7

Planning Standard 7 of 10 CFR § 50.47(b) requires that notifications be made periodically to the public setting forth how they would be notified in an emergency, and that contacts and procedures for the dissemination of information to the media be established. The record contains no particular criticism of media arrangements, and FEMA's 1983 Post Exercise Assessment concluded that media activities during the 1983 exercise functioned effectively. (Findings 3.1-71 and 72) Such attention as this planning standard did receive during the hearings appears to have been reserved for the public information brochure, "Indian Point Emergency Planning and You." This brochure directs residents of the plume exposure pathway EPZ to turn to local EBS radio stations upon hearing warning sirens, and to follow instructions broadcast on EBS. (Finding 3.1-66) The brochures also identify each recipient's Emergency Response Planning Area (ERPA), bus pickup locations for transit-dependent persons, and evacuation routes. (Finding 3.1-67) The brochures also contain

information on the evacuation of schools, hospitals and other facilities, background information on radiation, and a postage prepaid postcard for persons who may require special arrangements. In addition, toll-free telephone numbers and addresses where additional information may be obtained are listed. (Id.)

Intervenors criticized the brochures for a number of reasons, including inadequate distribution and a lack of foreign language editions. The latter point is discussed in connection with Contention 4.7 below. As for distribution of the brochure, the record reflects that mailing omissions discovered after the 1982 distribution were addressed in the 1983 distribution. The brochures were revised between 1982 and 1983, and both FEMA and intervenor witnesses noted improvements in the 1983 edition. (Finding 3.1-66) With the concurrence of FEMA, distribution of the 1983 booklet was held up in Westchester County until after the March 9, 1983 exercise so that revisions to the Westchester plan could be included. FEMA also concurred in a decision to hold up distribution of brochures in Rockland County pending revision of the county's own plan. (Id.) Meanwhile, residents of these two counties will presumably rely upon information contained in the 1982 edition to the effect that they should turn to EBS radio stations upon hearing sirens. The delay in distribution of the brochure in two counties would

be more significant were it not for a number of other radiological emergency planning public information activities which have been taken in the Indian Point vicinity. These include the posting of signs and posters to inform transients, the placement of advertisements in local newspapers and on local radio stations, the development of inserts to be placed in telephone directories, a speaker program and meetings with community groups, visits by the public to the plants, and a program to reach non-English speaking persons. (Finding 3.1-68)

Intervenor criticism of the brochures consisted of a suggestion that different brochures be developed for different sub-groups within the EPZ. Another witness introduced a public opinion-type survey intended to demonstrate the public's familiarity with and attitudes towards radiological emergency planning. We find much of this testimony to be of limited value for our narrow purpose here of assessing whether Planning Standard 7 has been complied with. That standard does not require any particular level of public sophistication on emergency planning subjects, but rather that information about staying indoors and listening to radio broadcasts in the event of an emergency notification be conveyed to the public on a periodic basis. The record reflects that a good deal more than just this has now been done. Deficiencies in the public information program cited by

FEMA in its 1982 Post Exercise Assessment were deemed corrected by the time of FEMA's December 16, 1982 Update Report. (Finding 3.1-70) Based upon our review of the record as to the distribution of the brochures and the descriptions of the balance of the public information program, and also based upon our review of the brochures themselves (FOE Ex. 1; UCS/NYPIRG Ex. 14), the Board concludes that the planning for public information and education meet the requirements of the planning standard.

Planning Standard 8

Planning Standard 8 of 10 CFR § 50.47(b) calls for the provision and maintenance of adequate emergency facilities and equipment to support the response to an emergency. Each of the four counties within the Indian Point EPZ has an Emergency Operations Center (EOC). (Finding 3.1-81) There is also a State EOC in Albany, New York and a State District EOC in Poughkeepsie which coordinate state activities. (Id.) The licensees have established a near-site Emergency Operations Facility (EOF) at the Indian Point Service Center in Buchanan, New York approximately one-half mile from the site. (Finding 3.1-80) The EOF would house the licensees' Emergency Director as well as County, State and Federal representatives, and would be the facility from which the overall

emergency response effort would be coordinated. An alternate EOF is located in the Eastview Service Center, outside the EPZ. (Id.) Each of these facilities was visited by the Board and representatives of the parties during the course of the hearings. Each of these facilities is equipped with the communications equipment referred to in our discussions of Planning Standards 5 and 6, above. As part of the licensees' onsite plans, Technical Support Centers and Operational Support Centers have also been established.

Both licensees also have multiple systems for onsite and offsite radiation monitoring. Within the plant site, each unit is equipped with survey instruments, laboratory instruments, personnel dosimetry (pocket ion chambers, thermoluminescent dosimetry and film badges), area monitors, and process radiation monitors. (Finding 3.1-205) Offsite, licensees have installed a 16 unit Reuter-Stokes monitoring system at various locations within two miles of the site, and Ludlum area radiation monitors are maintained at 25 offsite locations. (Findings 3.1-89 and 3.1-206) The licensees also maintain an extensive meteorological program, consisting of two towers and associated equipment which have been instrumented in accordance with NRC Reg. Guide 1.23 and NUREG-0654. (Findings 3.1-90 and 3.1-207) The meteorological and radiological data are given on a real-time basis to the

Meteorological Information and Dose Assessment (MIDAS) computer system in the EOF, where it can be instantaneously transmitted to MIDAS computer terminals in the State and county EOCs. (Id.)

The record indicates that much of the counties' radiological emergency response equipment has been supplied by the licensees under the Chapter 708 law referred to earlier, and also by the $210,000 payments made to the counties by the State pursuant to a voluntary agreement whereby the licensees paid this amount to the State for the counties' benefit. (Finding 3.1-85) Certain county witnesses testified that they had insufficient equipment, such as vehicles (J. McGuire, Tr. 3815), to implement the plan in their county. Each such witness also indicated that a lack of equipment which would normally be used in any emergency had not hampered their response to prior non-radiological emergencies. Other county witnesses testified that they had received from licensees all of the equipment they had ever asked for. (Id.)

The Board views claims of insufficient equipment for generic emergency response foreclosed by the Commission's ruling in the San Onofre case which we discuss above. Claims of a lack of equipment of a radiological nature are quite different, however. If this and other related planning standards are to have meaning, certain amounts of resources which would only be called upon in a

radiological emergency must be available to achieve compliance with the standard. Our search of the record here for claims of insufficient equipment of a specific radiological nature reveals that these are confined to concerns about adequate dosimeters and related radiological exposure equipment. These are discussed more fully in connection with Planning Standard 11 below, however it appears to the Board that concerns expressed at an early point in the hearings were ameliorated in testimony offered subsequently by the State establishing that sufficient numbers of dosimeters were then available.

FEMA's 1983 Post Exercise Assessment found that each of the deviations from the NUREG-0654 criteria applicable to this planning standard which FEMA had cited in its 1982 Post Exercise Assessment had been corrected. (Finding 3.1-82) No new deficiencies were cited following the 1983 exercise. The Staff has likewise found that the licensees' plans comply with Planning Standard 8 of 10 CFR § 50.47(b) as well as those parts of Appendix E that relate to it. (Finding 3.1-209) Based upon our review of relevant evidence in the record, the Board concludes that the requirements of this planning standard are satisfied.

Planning Standard 9

Planning Standard 9 of 10 CFR § 50.47(b) requires that there be adequate capability for the monitoring and assessment of the actual or potential offsite consequences of a radiological release. The meteorological and radiological equipment referred to above give the licensees substantial radiation monitoring and dose assessment capabilities. In addition to fixed-site monitors, the licensees have over 50 mobile dose assessment and field monitoring teams. (Finding 3.1-91) There is also independent and redundant radiological monitoring capability in each of the four surrounding counties, as well as State and Federal Intra Agency Radiological Assistance Program capability. (Id.) The MIDAS system referred to above as well as the Atmospheric Release Advisory Capability (ARAC) system permit the rapid prediction of plume movement and dose assessment, and all such data can be independently accessed by county and State officials. (Findings 3.1-89 and 3.1-211) Both the NRC and FEMA concur that the licensees' plans are in compliance with this planning standard. (Findings 3.1-92 and 212) The 1983 FEMA Post Exercise Assessment found that accident assessment functions conducted at the EOF and in Westchester, Rockland and Orange Counties were good.

Putnam County was rated fair. (Id.) The intervenors and interested states failed to produce evidence to the contrary. The Board finds based upon its review of the relevant evidence in the record that the requirements of Planning Standard 9 are met.

Planning Standard 10

Planning Standard 10 of CFR § 50.47(b) requires that a range of protective actions be developed for emergency workers and the public, that guidelines for selecting between protective action alternatives are in place, and that appropriate protective actions in the ingestion pathway EPZ (10 to 50 mile annulus) have been developed. Procedures to insure the protection of both onsite and offsite emergency workers are enumerated in the onsite and offsite plans. The State RERP and each of the four County RERPs provide for a range of protective actions to be taken within the plume exposure pathway EPZ and detailed guidance to be used in determining which action is appropriate under particular circumstances. (Finding 3.1-95) In addition, protective action options and guidance are provided regarding the ingestion pathway EPZ. (Id.)

In reaching a decision as to which protective action to implement the RERPs adopt and incorporate the

Protective Action Guides of the Environmental Protection Agency. Plant conditions, meteorological data, projected doses, and time frames for the implementation of different protective action and for the termination of the incident are indicated as factors to be considered in selecting appropriate options. (Id.) Protective action options within the plume exposure pathway EPZ are: 1) initial precautionary operations; 2) selective sheltering; 3) general sheltering; 4) selection evacuation; and 5) general evacuation. (Finding 3.1-97) The plans provide flexibility for implementing various protective actions for different populations within the plume explosure pathway EPZ, and for changing protective actions as circumstances may require. It was also undisputed that the licensees' plans provided adequate protective actions for onsite personnel. (Finding 3.1-218)

The aspect of this planning standard which received the most attention from the parties was evacuation planning. However, our review of the record indicates that sheltering would be the preferred protective action in a good number of scenarios, such as a puff release and in a fast-moving accident situation. (Finding 3.1-99) The evacuation portions of the county RERPs were initially prepared by a civil engineering firm selected by a committee with county membership. (Findings 3.3-6 and 7) We discuss the provisions of the RERPs relating to evacua-

tion in connection with Contentions 3.3, 3.6 and 3.9, below.

One aspect of the evacuation portion of the county plans is particularly germaine to our consideration of planning standard compliance here. This relates to the determination of the portion of the plume exposure pathway EPZ general population that would be transit-dependent if an evacuation were to be ordered. Using conservative assumptions, it was estimated that from 11 to 22 percent of the EPZ population would require public transportation in an evacuation. (Finding 3.3-11) The consultants did not model ridesharing, whereby evacuating persons would offer rides to evacuating neighbors, family members and co-workers. (Findings 3.1-110 and 3.3-20) The record indicates that 75 per cent of EPZ residents would be willing to give rides to others, and if just half of those indicating such a willingness were actually to give rides to others, the need for buses would be eliminated altogether. (Id.)

Perhaps the most germaine data on transit-dependency and bus demand derives from a survey performed by a professional pollster retained by intervenors. His testimony based upon his survey results indicates that only three per cent of the Westchester County EPZ population believe that they would require bus transportation in the event of an evacuation. (Id.) Based on our review of the

record, the Board finds that the present plans overstate transit-dependency among the general public within the EPZ by a significant amount, and thus the demand for buses in also overstated proportionately.

Bus evacuation capability is particularly significant due to the fact that in its testimony before the Board FEMA consistently characterized the bus component of the Westchester County RERP as significantly deficient under Planning Standard 10. (Finding 3.1-110) The origins of the concern over the bus situation in Westchester County appear to stem from doubts expressed by a former County official as to whether bus drivers would respond to evacuate the general population after an initial evacuation of school children. (Finding 3.1-108) This arose because the original Westchester County RERP provided for a "two wave" evacuation, with buses returning to the EPZ for a second trip in certain areas. However, the need for a "second wave," and we therefore assume the original basis for concern, has been mooted by changes in planning which were exercised during the March 9, 198[illegible] exercise. Under the new procedure, which has variously been referred to as the "early school dismissal plan" and the "O'Rourke plan," schools would be dismissed using generic state law "snow day" procedures at the alert stage of an incident at Indian Point. (Finding 3.1-108)

Since the Board is of the view that accommodation for ride sharing and revisions of the Westchester County plan to eliminate second wave evacuation would normally obviate further concerns over this planning standard, it is necessary for us to give close scrutiny to the position of FEMA. During the March 9, 1983 exercise the FEMA team leader in Westchester County who was responsible for evaluating the exercise there found that an "acceptable" ability to implement actions to protect the public (which includes evacuation) was demonstrated. (Finding 3.1-107) He also found that given the requirements of the exercise scenario, evacuation and transportation actions were "adequate." (Id.) However, in the letter accompanying the 1983 Post Exercise Assessment the evaluation of that portion of the exercise in Westchester County was revised to "significantly deficient." (Finding 3.1-110) The Board concludes that the apparent change in FEMA's assessment was not satisfactorily explained.

The testimony given by FEMA after the 1983 exercise suggests that its principal concern was the possible non-response of bus drivers. (Id.) The only other witnesses to testify on the subject of buses were the owner of one of the bus companies which would be involved in Westchester county, a bus driver, a union official, and an emergency planner in an adjoining county. The bus company owner testified that he believed most of his

drivers would perform their duties in an emergency. (Finding 3.1-109) The bus driver testified that five of six drivers in her garage who were sampled stated that they would respond. (Id.) The union official had no position, and the witness from the adjoining county testified that 88 of 100 bus drivers sampled indicated that they would respond in a radiological emergency. (Id.)

There appears to be no uniform FEMA policy regarding the extent to which assurances of the availability of transportation services for the transit dependent will be required for plan approval. FEMA Region 9 has found the transportation component of an emergency plan to be acceptable despite the absence of agreements with bus companies. (Finding 3.1-111) In the case of Indian Point, FEMA Region II did not receive any guidance from FEMA's national office on how to deal with the lack of written agreements. (Id.) At an earlier stage, FEMA rejected the possibility of using National Guardsmen as substitutes for Westchester commercial bus drivers who failed to respond to a radiological emergency because FEMA concluded that in a worst case situation the 4 to 5 hours needed to have the Guard in place would be inadequate. (Finding 3.1-112) FEMA has told the Commission that concerns about a "fast moving accident" led to the rejection of the National Guard option even though the additional time needed to mobilize the Guard would not be a concern in "other scenarios." (Id.)

In making its finding with regard to the significance of the lack of commitments from Westchester County bus operators FEMA did not attempt to determine significance in terms of public health, and did not consult with the NRC in making this finding. (Finding 3.1-113) No attempt was made by FEMA to assess the impact of this deficiency on evacuation time estimates. (Id.) FEMA agreed that in the face of a fast-moving type of accident where National Guard or other arrangements would not be feasible, sheltering and not evacuation would be the preferred protective action, and that in a sheltering followed by relocation scenario the urgency for bus transportation would be substantially reduced. (Finding 3.1-114)

The Board concludes that it cannot sustain FEMA's position that there is a significant deficiency under the applicable planning standard due to a possibility of non-response by bus drivers in Westchester County. We can find no basis in the record for concluding that bus driver availability is any less certain in Westchester County than in other counties or at other sites. The only testimony before us directly bearing on anticipated bus driver conduct uniformly suggests that a high proportion of bus drivers would respond. Our confidence that sufficient transportation resources would be available if needed in Westchester County is strengthened by: (1) our conclusion

that in failing to model ride sharing, the RERPs have significantly overstated bus demand, and (2) the elimination of the two wave evacuation concept. The Board accordingly finds that upon review of the evidence in the record, the requirements of Planning Standard 10 are met.

Planning Standard 11

Planning Standard 11 of 10 CFR § 50.47(b) requires the establishment of means to control radiological exposure to emergency workers consistent with EPA Protective Action Guides. The onsite and offsite emergency plans for Indian Point have detailed procedures for controlling and minimizing radiological exposure of emergency workers which are consistent with EPA guidelines. (Finding 3.1-118) Testimony offered by New York State indicated that 950 self-reading dosimeters as well as 5500 permanent record dosimeters (TLDs) had been distributed to the four counties as of the time of the preparation of the State's supplemental testimony. (Finding 3.1-119) There are also hundreds of civil defense dosimeters kept at emergency preparedness offices in each of the four counties, and the State has over 180,000 dosimeters stored by the State's Office of Disaster Preparedness. (Finding 3.1-120) FEMA has approved the use of civil defense dosimeters for radiological emergencies while more advanced low-level

self-reading dosimeters are phased in. (Id.) Although a number of witnesses contended that there were not enough low-range dosimeters in the hands of the agencies which they represent, none of the these witnesses were able to demonstrate why the approach adopted by REPG and FEMA (i.e. using a mix of different types of dosimeters while additional stocks are purchased) should be proscribed by applicable federal planning standards. (Finding 3.1-121) No significant deficiencies in radiological exposure control for emergency workers were found in the 1983 exercise. (Finding 3.1-123) Based on our review of the record the Board finds that the requirements of this planning standard have been met.

Planning Standard 12

Planning Standard 12 of 10 CFR § 50.47(b) requires that arrangements be made for contaminated injured individuals. Until the Commission's ruling in San Onofre, supra, there was arguably some doubt as to how this requirement should be construed. However the decision in that case makes clear that the planning standard refers to persons who are both injured and contaminated, and requires that emergency plans provide information with regard to the capabilities of medical facilities to treat such individuals. With regard to individuals who may be exposed

to dangerous levels of radiation, the Commission found that care could be provided on an as needed basis during an emergency, since persons in this group are unlikely to need immediate medical care. (Id., CLI-83-10, pp. 2-3) Measured by these standards, the emergency plans for Indian Point meet the requirements.

Licensees have entered into an agreement with Peekskill community Hospital to accept injured and contaminated personnel from the site. (Finding 3.1-127) The personnel at the hospital are regularly trained in techniques of dealing with injured and contaminated patients. Northern Westchester Hospital Center has agreed to serve as a backup hospital. In addition Con Edison has retained a medical consultant from Brookhaven National Laboratory Hospital to supervise special cases. (Id.) In addition to the hospitals with which licensee have agreements, the State RERP has identified 47 hospitals throughout the state, many of them within close proximity to Indian Point, which have indicated that they have the capability for treatment of contaminated individuals. (Id.)

An intervenor witness offered testimony on the results of a survey he conducted of hospital and ambulance services in the plume exposure pathway EPZ, however this testimony was given prior to San Onofre and proceeded on the assumption that the plans must provide for medical services for large numbers of the general public. While

there appear to be flaws in the survey offered by intervenors, we need not address them since the Board finds the study is premised on a reading of Planning Standard 12 which is inconsistent with the interpretation reached by the Commission. FEMA's 1983 Post Exercise Assessment found that simulations to test compliance with this standard went well. (Finding 3.1-128) Staff concluded that licensees' emergency plans are in compliance. (Finding 3.1-222) The Board finds that the requirements of Planning Standard 12 have been met.

Planning Standard 13

Planning Standard 13 of 10 CFR § 50.47(b) requires the development of general plans for recovery and reentry. Recovery and reentry procedures are detailed in both the State and county RERPs and the licensees' onsite plans. (Findings 3.1-133 and 3.1-228) Recovery and reentry procedures were tested during the March 9, 1983 exercise and found adequate by FEMA, and the Staff has also concluded that licensees' plans comply with this standard as well as the companion requirements of Appendix E.IV.H of 10 CFR Part 50. (Findings 3.1-134 and 3.1-229) In light of the record before us and in the absence of testimony alleging noncompliance, the Board finds that this standard is met.

Planning Standard 14

Planning Standard 14 of 10 CFR § 50.47(b) requires that periodic exercises and drills be conducted to evaluate emergency response capabilities, and that deficiencies identified as a result of exercises or drills are or will be corrected. The RERPs of the four counties, State and licensees provide for an annual exercise to test capabilities to respond to an accident. (Finding 3.1-138) Two such exercises have been conducted, on March 3, 1982 and March 9, 1983. The 1983 Post Exercise Assessment commented favorably on the relevance of the exercise, and as is clear from FEMA's December 16, 1982 Update Report and Table I of the 1983 Assessment, the respective parties have demonstrated a willingness to undertake extensive efforts to correct deficiencies. The Staff concludes that licensees comply with this standard. (Finding 3.1-236) The Board finds that the respective plans provide for the required exercises and drills, that an ability to implement corrective actions following exercises and drills has been shown, and that accordingly, the requirements of this standard are met.

Planning Standard 15

Planning Standard 15 of 10 CFR § 50.47(b) requires that radiological emergency preparedness training be provided for those who may be called on to respond in an emergency. For the most part the RERPs rely upon emergency response personnel to perform the same duties during a radiological emergency which they perform during a non-radiological emergency. (Finding 3.1-143) Thus policemen and others with responsibilities under the RERPs perform duties that are, for the most part, the same duties they are prepared to perform on a day-to-day basis, bus driver drive buses, the Red Cross operates host facilities and the like. In the 1981 and 1982 period licensees and consultants retained by licensees undertook a comprehensive training program of off-site personnel with emergency response duties in order to familiarize them with those portions of their duties which are unique to radiological emergency response. (Finding 3.1-144) Since that time, responsibility for training of emergency workers has been assumed by the State. Testimony offered by the State indicates that through February 1983, State-sponsored training had been given to more than 4800 persons. (Id.) The record reflects that the State and licensees have regularly been willing to provide radiological emergency response training on an ad hoc basis as well. (Id.)

Although much has been done to provide radiological emergency training, not all potential emergency workers had yet received training at the time the record of this proceeding closed. In Rockland County the pace of this training was adversely affected by the resistance on the part of the potential recipients to this training.

Nonetheless, training is an ongoing process, and the Board has been influenced by the substantial number of persons who have already been trained. The State testified without contradiction that the vast majority of police officers within the Indian Point plume exposure pathway EPZ have been trained in radiological emergency procedures. (Id.) The FEMA 1983 Post Exercise Assessment di not indicate any deviation from NUREG-0654 criteria arising under this planning standard, although there were recommendations for additional training in particular areas. (Finding 3.1-146) With respect to training for onsite workers, each licensee has had an extensive training program for many years, and the Staff has testified that these programs meet regulatory requirements. (Findings 3.1-238 and 3.1-240) Based on our review of the overall record as it relates to training, the Board finds that the requirements of this standard have been met.

Planning Standard 16

Planning Standard 16 of 10 CFR § 50.47(b) requires the establishment of responsibilities for plan development, distribution, review, and training. In New York State, the State's REPC has been given overall responsibility for radiological planning in the State, including liaision with other state agencies, local governments, federal agencies and the licensees. Its officials, who appeared before the Board, are well qualified to perform their responsibilities with the REPC. (Finding 3.1-149) Similarly, the various county personnel with responsibility for planning for radiological emergencies are well trained to perform their responsibilities. (Id.) Onsite compliance with this planning standard was testified to by licensees and Staff and was not controverted. (Finding 3.1-246)

FEMA has, since its July 30, 1982 Interim Findings, found compliance with this planning standard to be significantly deficient, based entirely upon a resolution by the Rockland County Legislature in May 1982 to withdraw from the coordinated planning process with the other three counties and the State. The Rockland resolution provided that the County would no longer participate with FEMA, NRC or licensees in the development of a federally mandated RERP and would prepare its own General Disaster Preparedness Plan. (Finding 3.1-150) The

Resolution provided, however, that, in the event of an actual radiological emergency, the County would cooperate fully with all agencies. (Id.)

The Board believes that the Rockland County leadership has not acted in the best interests of its citizens by withdrawing from coordinated planning. While County officials who testified before us contended that the County is working regularly on its own plan, there has been steady slippage in the County's timetable for completion. (Finding 3.1-159) If County officials were serious about protecting their constituents they would be following a more expeditious course in emergency planning. It appears that the County's position on emergency planning is motivated largely by a desire to force a shutdown of the plant.

In response to FEMA's concerns related to the Rockland County issue, New York State decided to modify the State RERP to provide for State compensating measures if the County failed to respond to an emergency. (Finding 3.1-154) The State RERP was in fact modified to provide for this possibility on a generic basis. On the basis of this modification of the State plan to provide for State fulfillment of any required protective actions which could not be fulfilled by county or local governments, the New York State REPG found that this planning standard is met at Indian Point. (Findings 3.1-29 and 3.1-154)

In its December 16, 1982 Update Report FEMA concluded that there remained a significant deficiency in this planning standard due to nonparticipation of Rockland County. FEMA has maintained this position after the 1983 exercise, in which Rockland officials participated until 9 A.M., and were then supplanted by State personnel. (Finding 3.1-155) FEMA has not to date evaluated the adequacy of the State's compensating measures program for Rockland county. FEMA stated in its Update Report that despite "significant progress" in Rockland county relating to funding, equipment, and training, the absence of a County plan precluded any finding other than a deficiency. The Report stated that "[u]ntil Rockland County's plan is developed, reviewed by the Regional Assistance Committee and personnel trained on the new plan, the original emergency plan must be utilized. This plan has been disavowed by the Legislature of Rockland County. The Chairman of the County Legislature has also expressed to the NRC that the assumption of County functions by State personnel would not be adequate." FEMA thus appears to have concluded that compliance with this planning standard could be had only if Rockland were to develop a plan of its own. (Finding 3.1-156)

This interpretation appears to be confirmed by FEMA's response to the Rockland County situation following the 1983 exercise. Despite a generally favorable review of

how the March 9, 1983 emergency exercise was actually conducted in Rockland County, the 1983 Post Exercise Assessment found the response significantly deficient.* FEMA reached this conclusion for two reasons. First, the county had not finalized and adopted a RERP and did not participate in the exercise. FEMA cited its proposed rule, 44 CFR Part 50, for the proposition that the concept of Emergency Planning Zones implies mutually supportive efforts by federal, state and local governments. (1983 Post-Exercise Assessment, p. 35) Thus, it appears that FEMA assumed that participation in emergency planning by all levels of government was required. Second, FEMA concluded that since the State RERP contemplated the state supplementing county personnel and resources, rather than supplanting them, the State had failed to follow its own plan during the exercise. (Id.)

The Board finds that because of the limited purposes served by the 1983 Post Exercise Assessment -- to evaluate the March 9, 1983 exercise -- it cannot be automatically used for the broader purposes required of

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* FEMA made no connection between its characterization of this deficiency as "significant" and increased risk to the public. (See Tr. 14831-32) Thus FEMA did not intend to convey in its 1983 Post Exercise Assessment that the two deficiencies labeled as "significant" would be at all significant in terms of their impact on the public in the event of an accident.

this Board: to evaluate compliance with Commission planning standards. We conclude that the planning standards should be interpreted in a functional way, asking whether in the event of an actual emergency the requirements of each standard would be sufficiently met so that harm to the public would be minimized. This requires us to look to a vital distinction between exercises -- where Rockland County has to date declined to participate -- and what would happen in an actual emergency -- where Rockland County would participate and cooperate fully.

In the absence of exercise experience the Board has looked to the record for other reliable indicators of Rockland County's ability to respond effectively to an actual emergency. The record indicates that over 1000 persons have received State-sponsored radiological emergency response training in Rockland County (New York State Ex. 9), and that the County's ability to respond to non-radiological accidents has been good. We also infer from Rockland County's failure to produce its chief radiological emergency planner to testify, as requested by the Board, that Rockland County could have demonstrated an adequate response capability during the March 9 exercise had it participated.

We also find that a functional assessment of the State's capabilities to "fill in" for nonparticipating county personnel requires us to look behind the FEMA

analysis that the State's compensatory measures program was insufficiently set out in written plan documents. A review of the 1983 Post Exercise Assessment indicates that the actual tasks performed by State personnel on March 9,1983 on behalf of Rockland County were carried out with FEMA's general approval. (Finding 3.1-155) FEMA's favorable assessment of the performance of the State compensating team during the March 1983 exercise confirmed the capability of the State to implement its compensating measures plan. We find ample support in the record, including the affirmative findings of the State's REPC, that the State plan provides reasonable assurance that Rockland County's portion of the plume EPZ would be protected in the event of an actual emergency.

Based upon the Board's foregoing evaluation, we find that the requirements of this planning standard have been satisfied.

Conclusion

Onsite and offsite emergency planning at Indian Point satisfy the planning standards of 10 CFR § 50.47(b) and Part 50, Appendix E.

HUMAN RESPONSE FACTORS

Implicit in any plan responding to potentially large scale accidents are assumptions about human behavior

in response both to the accident and to the plan for coping with it. By Contention 3.2, intervenors claimed that the Indian Point plans are deficient because they fail to take account of human response factors. It is asserted that the feasibility of timely evacuation would be affected.

Licensees contend that the plans are based on assumptions about human response that have proven correct in virtually all reported emergency situations, and that the evacuation time estimates and the feasibility of timely evacuation are correct. FEMA believes that the public would behave appropriately in the event of a radiological emergency at Indian Point. (Finding 3.2-14)

In formulating Contention 3.2, the Board intended that testimony should not be generic, but that it should focus on how the high population density in the vicinity of the Indian Point plants might exacerbate consequences resulting from a radiological emergency. We thus allow a narrow swath for this contention consistent with our direction that consideration of psychological impacts is not within the intended scope of this proceeding. In this regard, the parties offered no evidence that persons in the Indian Point vicinity are any different psychologically than those elsewhere. In fact, the theories offered by various intervenor experts -- if true -- would apply to every nuclear plant site. Again, the Board is precluded from examining such generic issues in this proceeding.

A major thrust of intervenor testimony under this contention was that the population would behave differently in a radiological emergency than in a non-radiological one. However, the record shows that the only research on point, conducted at Three Mile Island, shows that there is no material difference between response to radiological and non-radiological emergencies. (Findings 3.2-3 and 3.2-14) This is because the response is to the stress caused by an emergency, rather than to the emergency itself. (Finding 3.2-3) Testimony was also offered regarding the existence of "radiation response syndrome," allegedly a species of post-traumatic stress syndrome, which it was claimed might have an adverse effect on evacuation response. The Board finds, however, that even if radiation response syndrome exists, it is unlikely to be a significant factor in emergency response, since its proponents acknowledge that it would manifest itself only months or years after exposure to radiation. (Finding 3.2-6) Various predictions of poor public response to a civil radiological emergency were also based principally on the response to the atomic bombings of Hiroshima and Nagasaki. However, that experience was very different from a power plant accident, since it involved widespread and immediate physical destruction. (Finding 3.2-7)

Another major emphasis in the testimony was upon the response of emergency workers to a radiological

accident. The empirical evidence supports the conclusion that emergency workers would not abandon their posts as some witnesses suggested. Emergency workers in fact entered the TMI and Ginna plants after accidents, and there was unanimity of agreement among expert and non-expert witnesses at the hearings, and the Board so finds, that emergency workers have never failed to respond to emergencies in the past. (Finding 3.2-9)

We also find, based on the weight of the testimony, that special contracts or advance compensation arrangements are not necessary in order to ensure that emergency response personnel would respond if needed during a radiological incident. This is because emergency workers are not motivated by contractual commitments or the knowledge that they will be paid at overtime rates, but by the desire for crisis mastery and the knowledge that one is doing useful work rather than being a victim. (Findings 3.2-9 and 10)

With respect to the assertion that the general public might not respond in an orderly way to a directive for a mass evacuation, we simply find that the better view, based on historical experience, is that this would not occur. In the two most apt examples, Three Mile Island and an invisible chemical spill in Mississauga, Ontario the evidence showed that both emergency workers and the public responded in a timely, orderly, and obedient fashion.

(Findings 3.2-9 and 3.2-31) FEMA, which is well experienced in the area of emergency response, believes it can be safely assumed that individuals will behave appropriately in a radiological emergency. (Findings 3.2-9, 3.2-11 and 3.2-14) Instances of panic in emergency situations have been confined to incidents, such as building fires, in which egress is blocked or constricted in an enclosed area, which would not be the case in a mass evacuation. (Finding 3.2-14) Also, attitude surveys about how people would respond to an emergency are not good predictors of actual emergency behavior. (Finding 3.2-18)

Another issue addressed in the testimony is spontaneous or "shadow" evacuation, which it was claimed would occur prior to the issuance of an evacuation order. The Board cannot find a coherent argument for why this would adversely impact an effective response. To the extent spontaneous evacuation were to occur, it would reduce evacuation times, not increase them. Any movement of population out of the EPZ before the evacuation order would mean that there are fewer people in the EPZ than modeled in the evacuation time estimates. (Finding 3.2-16)

Conclusion

The evidence taken as a whole is persuasive that the past responsible behavior of emergency workers to non-radiological emergencies would also hold true for

radiological emergencies, that the response of the general public to a radiological emergency would not be materially dissimilar to that observed in past mass evacuations, that in particular, there is no reason to assume that panic would occur, and that pre-announcement self-evacuation would ease, not hamper, the response. Consequently, the Board finds that assumptions made in the Indian Point emergency plans with respect to human response factors during a radiological emergency are not erroneous, and that the estimates of evacuation times and of the feasibility of timely evacuation for certain areas are not made incorrect by use of such assumptions.

THE RELIABILITY OF THE INDIAN POINT EVACUATION TIME ESTIMATES

Even prior to the amendments to the Commission's emergency planning regulations in 1980, the Indian Point licensees had evacuation time estimates prepared for the site's plume exposure pathway. These estimates were challenged by intervenors in Contention 3.3, which asserted that the estimates are unreliable, based on unproven assumptions, and utilize unverified methodologies.

The evacuation time estimates are set forth in the evacuation portion of the State and county RERPs. They were prepared by a civil engineering firm experienced in traffic engineering work, under contract to the licensees.

The consultant firm which prepared the evacuation time estimates also prepared two lengthy documents setting forth the methodology and data which they used in preparing the estimates, as well as a guideline document for updating the estimates. (Finding 3.3-4)

In preparing the evacuation time estimates the consultant worked closely with county and State officials. (Finding 3.3-6) They broke down the plume exposure pathway EPZ into 46 Emergency Response Planning Areas (ERPAs), based on a careful study of the EPZ. (Finding 3.3-7) For purposes of preparing the estimates, persons to be evacuated were classified either as members of the residential population, special facilities population or transient population. Residential population was identified using data consisting primarily of estimates of the 1980 population which were prepared by each of the four counties located in the EPZ. (Id.) The category of special facilities includes: schools, colleges, nursing homes, hosptials and other health care facilities, convents and monasteries, group homes, correction facilities, and camps. These facilities were inventoried through directories, letters and by telephone. The transient population includes people employed within the EPZ, people staying at hotels and motels in the EPZ, and visitors to parks and recreation areas. (Id.) Six scenarios were considered for purposes of preparing estimates and a

different time period determined for each scenario. The scenarios consisted of night, weekday/school in session, weekday/ school not in session, evening, weekend or holiday/summer daytime, and weekend or holiday/winter daytime. (Finding 3.3-8)

The methodology used by the consultant for calculating the evacuation time estimates embodied in the Indian Point emergency plans was reviewed and approved by the New York State Department of Health, the New York State Police and the New York State Radiological Emergency Preparedness Group (REPG). (Finding 3.3-9) The roadway portion of the analysis received the closest scrutiny in the hearings. The roadway evacuation capacity analysis involved a survey of primary and secondary evacuation routes. Every primary and secondary route was traveled by consultant's personnel, and primary routes were also driven by a FEMA expert who testified before the Board. (Finding 3.3-12) Roads were examined outside of the EPZ, to the next major intersection, to ensure that no bottlenecks outside the EPZ could cause traffic to backup into the EPZ. (Id.) Capacities were calculated for each link in normal and adverse weather conditions. The estimated capacities were validated by comparison with New York State Department of Transportation data. (Finding 3.3-12)

The consultant determined road capacities by use of the Highway Capacity Manual, which is the standard reference work for such purposes. (Finding 3.3-13) Among the road features considered in determining the capacity of each road segment were road width, shoulders, and impact of traffic signals. The consultant assumed capacity restrictions on the roadway system to account for possible adverse weather conditions. (Finding 3.3-15) The evacuation network used for preparing the estimates is based wholly on the presently existing roads within the EPZ, and thus considers the principal roadways which would be used in an evacuation. (Finding 3.3-12) While "minor" roads are occasionally specified, the number of vehicles expected to travel such roads is similarly small.

The evacuation time estimates made conservative assumptions about the percentage of the evacuating population expected to be dependent on public transportation. It was assumed that persons without cars would not get rides from neighbors, although a survey of the area showed that 75% of those with cars would offer rides to neighbors without a car. (Finding 3.3-20) The evacuation time estimates assume an 11 to 22 percent transit-dependency. (Id.) However, a survey conducted by intervenors and accepted into the record determined that only 3% of the general public in one EPZ county would require public transportation in the event of an evacuation.

Indian Point EPZ evacuation time estimates were also prepared for FEMA by CONSAD Research Corporation. Those estimates produced results that consistently matched the consultant's estimates. What differences there were showed longer estimates for the consultant's work. FEMA's expert witness testified that he had examined the initial version of the consultant's work for the licensees and also CONSAD's work, and had rated them "excellent" in their conformance with the requirements of NUREC-0654. (Finding 3.3-29)

A large number of intervenor witnesses purported to address this contention, however their testimony generally asserted that the EPZ had numerous narrow and winding roads. The one witness with traffic engineering experience who testified on behalf of intervenors had not reviewed the methodology used by the consultant who had prepared the evacuation time estimates, and was unfamiliar with the evacuation time estimate guidance in NUREG-0654. (Finding 3.3-32)

Conclusion

In evaluating the evidence on evacuation time estimates the Board must remain cognizant of the purpose to which they are to be put. The purpose of creating evacuation travel time estimates, as laid out in Appendix 4 of NUREC-0654, is to aid State and local officials in

selection of an appropriate emergency response option in the event of an incident at Indian Point. Given the function of the evacuation time estimates, the Board concludes that they both satisfy the requirements of NUREG-0654 and provide the counties with an adequate framework for implementing an evacuation should one become necessary. The evacuation plan and procedures include the data necessary for emergency managers to make decisions concerning the selection of protective actions as well as the operational strategies for carrying out evacuation-related activities, and can be appropriately used by emergency response decision makers for these purposes. The Board further finds that the evacuation travel time estimates contained in the RERPs and companion documents are reasonable representations of the time which can be expected to be actually required to evacuate some or all of the EPZ. The estimates are based on accepted transportation engineering methods applied in a regular manner, and are a realistic analysis of the area's existing transportation facilities and population subgroups.

LICENSEES' ADMINISTRATIVE CONTROLS PERTAINING TO NOTIFICATION

In the event of an unusual event occurring at a nuclear power plant, generally the first offsite notification is made by the plant operating personnel. By

Contention 3.4, intervenors contended that the Indian Point licensees could not be depended upon to make prompt notification. The testimony indicates that the licensees have the capability to notify offsite authorities of plant conditions, as required by 10 CFR, part 50, Appendix E.IV.D.3 and NUREG-0654, Appendix 1. Staff testified that licensee personnel understand the notification procedures and their responsibility for implementing them, that the licensees have been conscientious about reporting incidents to the NRC, and that in fact Indian Point personnel report more quickly and more completely than at any other plant in the country. We have reviewed in our Findings the extensive record of notification to offsite State and local officials during emergency planning exercises and at other times, and the record shows that licensees report incidents which are not reportable under Commission regulations. (Finding 3.4-2) The general burden of the testimony is that administrative control of notification and reporting procedures at Indian Point have been substantially upgraded, to the point that the NRC Staff finds them exemplary. State, county and local governments expressed no disapproval of current reporting. (Finding 3.4-6) Such evidence as was offered by intervenors related to the level of public confidence in licensees, which is irrelevant to Contention 3.4 as formulated.

Conclusion

Based upon uncontroverted evidence, the Board finds that administrative controls and procedures at the Indian Point units are sufficient to give reasonable assurance that the licensees can be depended upon to notify the proper authorities of an emergency promptly and accurately enough to assure effective response.

CONSIDERATION OF METEOROLOCICAL CONDITIONS IN EMERCENCY PLANS

Emergency planners are required to take into account unusual weather conditions in formulating appropriate emergency responses. Contention 3.6 appears to allege that unusual weather conditions exist in the Indian Point area, for which adequate assessment and protective actions have been not planned. This position was disputed by licensees, the Staff, FEMA, the State and several county witnesses. The record reflects that the Indian Point licensees have devoted an unusual degree of attention to assessing site meteorological conditions. No witness was able to identify any severe weather conditions peculiar to this site. The record shows that the most severe snowstorm occurring at the most inopportune time would make EPZ arterial roadways impassible for a maximum of 4 to 6 hours. (Finding 3.6-6) We note that in the event weather conditions made evacuation inadvisable, the RERPs provide

for sheltering as an appropriate protective response. There was no evidence of unique meteorological conditions in the vicinity of Indian Point that would affect emergency planning in a way which was different from the consideration of meteorological factors at other nuclear power plants.

Conclusion

The uncontroverted testimony establishes that meteorological conditions have been appropriately taken into account in the current plans. The Board therefore finds that no case has been made for adoption of Contentions 3.6.

THE EVACUATION OF CHILDREN

Children would of course be evacuated along with other members of the public were a general evacuation to be ordered. In Contention 3.7, intervenors contended that problems associated with accomplishing the evacuation of children had not been satisfactorily addressed. The situation that generated most of the testimony arising under the Contention concerned evacuation while school was in session. Under the RERPs as formulated at the outset of the hearings, in the event of an evacuation school children were to be bused in the company of their teachers directly to reception centers, where they would be reunited with

their families. Intervenors sharply criticized these arrangements, asserting that some parents might try to pick up their children at the schools, thereby disrupting the plan.

As a result of this concern, the approach to planning has changed so that schools are to be dismissed and students sent home at an early stage of any radiological emergency, to evacuate along with their families or others. (Finding 3.7-6) The early school dismissal (or "O'Rourke") plan was tested in the March 9, 1983 exercise and found to be workable. (Finding 3.7-8) Both State and county officials testified that the revised procedures were believed adequate to provide reasonable protection to children. (Finding 3.7-6) The testimony indicates that the option to dismiss schools early, using generic "snow day" procedures required by State law, has always been available under the plans and has been used some 20 times in Orange County in the last five years, where it has proved workable. (Finding 3.7-7) An advantage of early dismissal is that it allows school bus drivers to drive their regular routes. (Id.) The New York Education Law requires each school district, and parents, to have pre-existing arrangements for go-home situations. (Id.) Each family is required to have on file with the school district instructions as to what to do when school is dismissed early and there is no adult at home. (Finding 3.7-9)

Children who may be returning to empty homes, or children at home without adult supervision, were referred to as "latch key" children in the hearings. Intervenors expressed concern about their well-being in the event of a radiological emergency. There was testimony offered that experience in other disasters has shown that individuals rely on the extended family and other important social contacts during warning and response periods in disasters, and that friends and relatives will assist special populations, such as latch-key children, in an emergency situation. (Finding 3.7-11)

The RERPs and emergency planning brochure include procedures available for such children, including bus pick up points and return postcards in the brochures for parents to indicate that children may be alone and could require assistance in a radiological emergency. (Finding 3.7-10) The testimony indicates that the plans provide for bus service for all transit-dependent persons, and that there are enough buses available to ensure evacuation of children in all counties of the EPZ. (Finding 3.7-13) We see no other reasonable planning measures that could be taken by responsible government officials.

Conclusion

The Board finds that implementation of the early school dismissal or O'Rourke plan overcomes objections

voiced earlier about the evacuation of children, and that this plan produces no insoluble problems with so-called latch-key children, for which all reasonable evacuation arrangements have been made.

ADEQUACY OF THE ROADWAY NETWORK

Intervenors claimed in Contention 3.9 that the roadway network in the vicinity of Indian Pint is inadequate for timely evacuation in the event of an incident. As we discussed in connection with Contention 3.3, the evacuation time estimates prepared for the RERPs to satisfy the requirements of the Commission's emergency planning regulations relied upon the roadway system as it presently exists, using thoroughly validated methodologies and conservative assumptions about traffic loads during an evacuation. (Finding 3.9-3) The estimates were validated by an independent study conducted for FEMA which produced similar results. The NRC/FEMA guidelines contain no time requirements for evacuations. Under the regulations there is no such thing as an evacuation time that is unacceptable so long as it is sufficiently accurate to fulfill its function of guiding decision makers.

While it is not relevant to regulatory compliance, there are useful insights which can be drawn from the risk portion of this proceeding into the timeliness of evacuation. As we discussed in connection with Commission

Question 1, the most probable accident scenario at Indian Point is a slowly evolving overpressurization failure of containment. In this sequence, containment failure would occur no sooner than 12 hours, and more likely more than 24 hours, after the accident began. (Finding 1.1-149) The slow development of this sequence would allow ample time to terminate the accident and to mobilize to implement evacuation. Indeed, most people could walk out of the EPZ if necessary. This scenario may entail evacuating substantial populations, however there would be ample time to do so. The longest evacuation time estimate determined for Indian Point (which would entail adverse conditions) is about 15 hours, and under most conditions evacuation time would be substantially less. (Finding 3.3-25) The only other accident scenario having a material contribution to risk at Indian Point is the interfacing systems LOCA, for which the preferred protective response is sheltering.

Conclusion

We have found the Indian Point evacuation time estimates to be accurate. There is no requirement under the Commission's regulations for evacuation in any particular time period. The evacuation time estimates have been prepared based on the existing roadway network. The Board therefore finds that no case has been made for adoption of Contention 3.9.

EVACUATION FOR THE MOBILITY IMPAIRED

Contention 3.10 is based upon allegations that adequate protection has not been given such groups as the physically handicapped, non-English speakers, persons with learning disabilities, the deaf or hearing impaired, senior citizens, "latch-key" children, special facility residents, retarded persons, and park visitors. The State and licensees, however, took the position that these approaches are reasonable. A review of the Indian Point emergency plans indicates that considerable attention has been given to the needs of these groups. The plans contain extensive identification of special facilities and, return postcards and toll-free telephone numbers contained in the emergency planning brochures serve to identify persons with special needs. (Finding 3.10-11)

Each county plan contains a Special Facilities Procedures section that addresses the evacuation of individuals from special facilities. (Finding 3.10-3) Evacuation steps for each type of facility are described. A table displays the map number on which each facility can be found, the ERPA number, the number of patients, a description of vehicles owned by the facility, a description of the number and type of vehicles needed for augmentation, and the telephone number of the facility. (Id.) All hospitals and nursing homes in the EPZ are

required to have evacuation plans, and there was testimony that it is doubtful any such institution has failed to make such plans. (Id.) The plans also provide special bus routes to pick up mobility-impaired transit-dependent persons. (Finding 3.10-7) FEMA testified that significant improvements in planning have or are scheduled to be undertaken for those ins itutions within the 10-mile EPZ that house mobility-impaired individuals. (Finding 3.10-9) The testimony before us indicates that the majority of special facilities own vehicles that would be used if evacuation became necessary. The plan identifies private bus companies that would be used to provide supplemental buses and vans. (Finding 3.10-10)

The county RERPs provide two primary methods for identifying the non-institutionalized mobility-impaired transit-dependent population: (a) the public information pamphlets distributed annually include a mail-in postal card, so that this population can be identified in advance, and (b) at the time of an accident, a special telephone number or numbers will be announced over the broadcast media for mobility-impaired persons to call to request transportation. (Finding 3.10-11) Each of the four counties in the EPZ has agreed to contact appropriate agencies and facilities to solicit further information about additional mobility-impaired persons. Also, in the event of an actual emergency, each county EOC would be

manned by a representative from the county agency responsible for the special problems of mobility-impaired persons. If evacuation becomes a possibility, the staff will activate its procedures, including notification to and coordination with EOC command.* (Finding 3.10-12)

These means are available not only to individuals with special needs themselves, but also to friends, relatives and organizations who can identify others with special needs. As a general proposition, members of virtually all of these groups cope with problems of transportation, notification or emergencies which are much more common than a radiological emergency. FEMA testified that most of the non-institutionalized mobility-impaired transit-dependent persons within the EPZ live with other individuals or have neighbors who would assist in protecting them in the event of an accident. Major responsbility for evacuation of mobility-impaired persons lies with the families and friends of those persons, in a radiological emergency just as in more frequently occurring situations. (Finding 3.10-14) As the State of New York and FEMA witnesses testified, these groups can and must, to

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* The policy of the New York State Department of Corrections is that inmates at the Ossining Correctional Facility will not be evacuated, but will instead be sheltered and provided with potassium iodide if necessary. The procedures of the Department of Corrections provide for staffing of the facility. (Finding 3.10-13)

a certain degree, help themselves by making their needs known to the proper authorities and by relying upon their friends and families to see that they are taken care of.

The intervenor witnesses who addressed this contention for the most part simply asserted that the plans were inadequate to assure the evacuation of mobility-impaired transit-dependent persons. The few references to the specific institutions or groups mentioned in the contentions merely stated that these institutions existed and stated that in the opinion of the witness these institutions could not be evacuted.

Conclusions

The record reveals that substantial planning has taken place for the protection of mobility-impaired persons in the Indian Point vicinity in the event of an incident, and that reasonable steps have been taken to identify such persons in advance. Testimony in support of the contention pointed to no deviations from regulatory guidance, and experts from FEMA, the State of New York, the NRC Staff, and the licensees all testified that the plans provide adequately for evacuation of such persons. The Board therefore finds that adequate measures specifically designed to protect persons with special needs have been taken.

QUADRANT EVACUATION UNDER COMMISSION QUESTION 3

The text of the Commission's question asks the Board to establish what the minimum number of hours warning for an effective evacuation of a 10 mile quadrant at Indian Point would be. We believe that the 90° quadrants evacuation travel time estimates set forth in the record as Con Edison Exhibit 10 constitute reasonable representations of what the travel times would likely be. We have accordingly included the quadrant evacuation travel times in our findings as Finding 3.11-3.

IMPROVEMENTS IN EMERGENCY PLANNING IN THE VICINITY OF INDIAN POINT

When the Commission issued its January 8, 1981 Order the revised emergency planning regulations had just been promulgated and State and local governments around the country were making preliminary plans to put the regulations into effect. The first half of Commission Question 4 asked the parties to supply information on what improvements could be expected in the level of emergency planning in the vicinity of Indian Point, and on what time schedule. It is a fair assumption that in formulating Question 4, the Commission envisioned the Board assembling a record during the period when activities to meet the new emergency planning regulations were in progress but in some respects remained incomplete. Notwithstanding the passage of time, this has to a great extent proven to be the case. Testimony and exhibits submitted by both the New York State REPG and FEMA emphasize in numerous ways that emergency planning is an ongoing process and can never be considered complete. Thus the State REPC testimony referred to training that is ongoing today, and successive FEMA documents such as the 1982 Post Exercise Assessment, the Update Report, and the 1983 Post Exercise Assessment each refer to continuing progress which is being made to further improve offsite emergency planning around the Indian Point site.

The second half of Commission Question 4 offered opportunities for creativity to the parties to the proceeding to suggest other offsite emergency planning procedures. The opportunity was not unlimited, however. Since participants in NRC Licensing Board proceedings are rarely if ever permitted to argue for requirements at variance with Commission regulations, the Commission did impose conditions. In its formulation of Question 4, the Commission added two important requirements: that proposed emergency planning procedures be "specific," and also that they be "feasible." And in a subsequent Order the Commission elaborated on the requirements for the second half of Question 4 by stating that proposed new procedures should be "prudent risk-reduction measures in light of the risk posed by Indian Point as opposed to the spectrum of risks posed by other nuclear plants." (July 27 Order at 15-16)

With these requirements in mind, the Board admitted six contentions and subcontentions under Question 4. As we proceed to discuss them individually, we must emphasize that our evaluations are necessarily bound by the limitations referred to above which the Commission has imposed.

EXPANSION OF THE PLUME EXPOSURE PATHWAY EPZ

The plume exposure pathway EPZ is the area around nuclear power plants where comprehensive emergency planning is required by Commission regulation 10 CFR § 50.47(c)(2). That regulation specifies "an area about 10 miles in radius," but continues that the "exact size and configuration of the EPZs surrounding a particular nuclear power reactor shall be determined in relation to local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries." We admitted as Contention 4.1 an assertion that the Indian Point EPZ should be expanded from its present 10-mile radius for the reasons set forth in the regulation, i.e., local needs as affected by demography, topography, etc. The principal theory of the contention's sponsors, however, centered not on demography and the like, but on severe accident considerations. This in turn requires that we review some of the background of the selection of the 10-mile radius.

The basis for an EPZ of about 10 miles was an analysis published in 1978 entitled "Planning Bases for the Development of State and Local Government Response Plans in Support of Light Water Nuclear Reactors," (NUREG-0396). The analysis was not performed using plant-specific data,

but instead used accident consequence modeling results from the Reactor Safety Study or WASH-1400. The NUREC-0396 analysis produced two conclusions particularly relevant to this contention: (1) that for the worst core melt sequences, immediate life threatening doses would generally not occur outside the 10-mile zone; and (2) that detailed planning within a 10-mile radius would provide a substantial base for an ad hoc expansion of response efforts in the event that this proved necessary in the course of an accident.

Thus the rationale underlying selection of "an area about 10 miles in radius" as the zone for comprehensive emergency planning included encompassing the area where there was early fatality (immediate life threatening doses) risk, and the "springboard" potential for response efforts outside the zone if needed. However, when the same type of analysis as performed generically in NUREC-0396 based on WASH-1400 analysis was done on a plant and site-specific basis for Indian Point, the results indicate that most (about 95 percent) of the early fatality risk occurs within 4 miles of the Indian Point site, an area considerably smaller than the current EPZ. (Findings 1.1-274 and 4.1-9) While this may supply a basis for reducing the plume exposure pathway EPZ at Indian Point, on the record here there is no basis for expanding it using the factors which were considered important in initially

selecting the radius. With respect to the expansion of planning outside the EPZ as needed during an incident, we take notice that the greater portion of the territory of each of the four counties comprising the Indian Point EPZ is located outside the EPZ, thus the area for protective response could be more easily expanded in the event of an actual emergency since the same emergency managers and political subdivisions would be involved.

Two other topics relevant to EPZ size bear discussion. With respect to the local conditions and demographic, topographic and similar conditions referred to in the regulatins and in our Contention 4.1 as formulated, the record reflects that the EPZ boundary, which has a jagged edge and is generally more than 10 miles from the site, was derived by mutual agreement of the State, the licensees, each of the four counties, and consultants. (Finding 4.1-7) Testimony in the record indicates that the local factors referred to were fully considered. Intervenors' testimony did not refer to EPZ size from the standpoint of demography, topography and the like but was rather directed towards a substantial enlargement of the zone.

The remaining topic is meteorology. Site-specific meteorological data was used in the IPPSS, and thus the conclusion that early fatality risk is generally confined to distances substantially less than 10 miles from Indian

Point fully reflects local meteorological conditions. Expert meteorological testimony at the hearings established that the Indian Point vicinity experiences a valley wind system, and that combined meteorological and topographical conditions are such that plume transport beyond the current EPZ is highly unlikely. (Finding 4.1-10)

Conclusion

Expansion of the current Indian Point plume exposure pathway EPZ cannot be justified based upon considerations which were important in the selection of the about 10 mile radius specified in 10 CFR § 50.47(c)(2). The record before the Board indicates that the local conditions of topography and the like specified in the regulation were considered. We therefore conclude that no basis has been shown for expansion of the Indian Point EPZ. Given our decision on this contention, we need not reach the question of whether an expansion would be warranted under the principles of Southern California Edison Company (San Onofre Nuclear Generating Station Units 2 and 3), LBP-82-39, 15 NRC 1163 (1982).

DISTRIBUTION OF POTASSIUM IODIDE

Ingestion of potassium iodide (KI) results in its absorption by the thyroid gland, which would "block" subsequent absorption of radioactive iodine released in a

serious accident. By Contention 4.2(a), it was proposed that KI should be provided to all residents of the EPZ. The record shows that KI distribution is opposed by many public health experts such as the New York State Health Department and the New York Academy of Medicine because of side effects, possible increase in bone marrow dose, "panacea effect," and other reasons. (Finding 4.2(a)-6) The New York State Disaster Preparedness Commission opposes its distribution to the general public. (Id.) Extensive quantities of KI are stored at Indian Point and could easily be distributed to emergency workers on short notice in the event of an emergency if conditions warranted. The only intervenor witness favoring KI distribution had not examined the reasons for the positions taken by public health organizations. (Finding 4.2(a)-8) We can find no basis for rejecting that judgment.

SHELTERING CAPABILITY

A protective response option frequently overlooked in discussions of radiological emergency planning is sheltering. The record in this proceeding indicates that sheltering would be the preferred response in a number of instances. By Contention 4.2(b) it was proposed that adequate sheltering capability should be provided for all EPZ residents. Much of the evidence proferred under this contention reflects a substantial misconception about

sheltering from nuclear plant accidents. The record indicates that sheltering, in the context of a radiological emergency caused by a release from a nuclear power plant, does not require civil defense or nuclear attack-type shelters with blast and fallout protection. Instead, it merely requires going indoors and controlling ventilation, as by closing doors and windows and turning off air conditioners. (Finding 4.2(b)-6)

The resources necessary to implement sheltering are an alert notification system and an EBS system. Both systems were found to have worked well during the 1983 exercise. (Finding 4.2(b)-7) Intervenor testimony on sheltering consisted of conclusory assertions that houses in the Indian Point area were inadequate for sheltering. Both FEMA and State found this not to be the case. The State testified that the principal element of sheltering was the minimization of air exchange, and that houses in the Northeast were generally designed for airtightness to save on winter heating bills. Even if this were not so, however, the Board believes that before such a proposal could be entertained there should be some showing that sheltering capabilities are materially substandard at this site. No such showing was made, nor were "specific" or "feasible" proposals for improving sheltering advanced as required by Commission Question 4.

PLANT SHUTDOWN IN BAD WEATHER

Contention 4.2(c) suggested amendment of the operating licenses of the Indian Point units to prohibit operation when road conditions become degraded due to adverse weather conditions. At the outset, we note that this Contention does not suggest an offsite measure, and thus is not a proper contention under Commission Question 4. Nevertheless, since it was litigated at the hearings, we will address it here.

The record reflects no severe meteorological conditions at Indian Point. The record also contains uncontroverted testimony that even under severe winter storms, the roads would most likely be passable within a few hours. Snow would, in any event, reduce the likelihood of wide plume dispersion. (Finding 4.2(c)-8) While no significant reduction in risk accompanying adoption of this proposal was shown, there would be some risk increases associated with frequent unit startup and shutdown. We must therefore decline to adopt the contention.

IMPROVING THE ROADWAY NETWORK

Contention 4.2(d) calls for upgrading of roads to permit successful evacuation of all residents of the EPZ before plume arrivial. In view of the Commission's recent decision in the San Onofre case, discussed earlier, and the

Commission's Question 4 requirements, we doubt whether the massive construction project apparently proposed by this contention is a proper matter for consideration in this proceeding. However, we need not rely on these bases alone. The implicit premise of the contention is that a timely evacuation is not possible in the vicinity of this site. We have found the contrary in connection with Contentions 3.3 and 3.9. We depend there, as we do here, on FEMA's finding that roads are presently adequate for evacuation purposes, and on the agreement of officials of three of the four EPZ counties with this position. (Finding 4.2(d)-5) We therefore decline to adopt the contention.

IMPROVED ALERTINC FOR SPECIAL POPULATIONS

Contention 4.7 proposed upgrading emergency plans to more adequately alert the blind, the deaf, those too young to understand instructions and those who do not speak English. We note that Contention 4.7 as formulated involves the narrow issue of alerting and informing four defined groups. Under Commission Question 4 proponents of this contention were required to suggest specific measures to alleviate any problem and show that these measures were feasible.

The primary method of alerting persons in the EPZ is the siren system. The primary method of informing them

is the Emergency Broadcast System. Both systems have been shown to be in good operating order. Apart from mere alerting, as we discussed in the context of Contention 3.10, extensive procedures have been developed for these groups. It is difficult for the Board to conceive of additional feasible planning procedures that government could undertake. Members of these groups cope with notification, information, and emergency problems much more common than radiological accidents. Parents, friends, neighbors, and relative who usually care for them can help them plan for an emergency and could assist them if an accident actually occured.

With respect to those who do not speak English, there was of course no testimony that they cannot be alerted by sirens, and such persons could certainly receive interpretations of EBS instructions from bilingual relatives or friends. Intervenor assertions that there was a substantial population of non-English-speaking persons within the EPZ was based on census data that did not support their position. FEMA, which was initially concerned about the language problem, determined along with the State of New York that there was no substantial difficulty with alerting those within the Indian Point EPZ who do not speak English. (Finding 4.7-5)

We decline to adopt Contention 4.7. No feasible and specific additional measures were advanced on this topic which we could recommend to the Commission.

INDIAN POINT RISK IN COMPARISON TO OTHER NUCLEAR PLANTS

The Commission's fifth question asked how the risks posed by the Indian Point units compare with the range of risks posed by other nuclear power plants licensed to operate by the Commission. This topic had initially been addressed by an NRC Task Force formed by the Commission's Order of May 30, 1980 to preliminarily examine the risk of Indian Point compared to other plants. The inquiry was prompted at that time by a request from the Union of Concerned Scientists that the Indian Point units not be permitted to operate for reasons of excessive risk during the pendency of the hearings before the Board.

The Task Force's report, entitled "Report of the Task Force on Interim Operation of Indian Point" (NUREG-0715), was issued in June 1980. The Task Force considered various risk-related factors including comparative site demography, accident risk comparisons, and the effects of emergency response, and plotted early fatality, early illness, latent cancer and property damage risk curves for the Indian Point plant assumed to be sited at the Indian Point site as well as five other sites (NUREG-0715, Figs. 1-4), and the Indian Point plant and three other plants of different design all assumed to be sited at the Indian Point site (NUREC-0715, Figs. 7-10). Basing its assessments on risk and source term data then available, the Task Force's principal conclusion was that:

> "[T]he [Indian Point] site is about an order of magnitude more risky than a typical site and the design about as much less risky than a typical design. There is much more certainty in our comparison of the relative site risks than there is in the comparison of the design risks. It is reasonable to conclude that the two about cancel, that is, the overall risk of the Indian Point reactor is about the same as a typical reactor on a typical site." NUREG-0715

In its January 8, 1981 Order (13 NRC 1), the Commission recognized that the Task Force report had not "disclosed its detailed methodology and underlying data," and that it also had not been "tested in an adjudicatory setting." (13 NRC at 5) That same Commission Order by its Question 5 directed the Board to address comparative risk issues in the hearings. In fact, the January 8, 1981 Order stated that:

> "The Commission's primary concern is the extent to which the population around Indian Point affects the risk posed by Indian Point as compared to the spectrum of risks posed by other nuclear plants. The Commission is concerned with both the total risk to persons and property posed by the Indian Point plants and the risk to individuals living in the vicinity of the Indian Point site, including that resulting from the difficulty of evacuation in an emergency. The Commission intends to compare Indian Point to the spectrum of risks from other nuclear power plants, since the primary basis for the Commission's decision will be how extreme are the individual and societal risks associated with Indian Point compared to the spectrum of risks from other operating stations." 13 NRC

Since the issuance of the Task Force's report there have been a number of significant developments affecting PRA in general and the risk of the Indian Point plant in particular. Not the least of these was the

completion of IPPSS, which we have discussed in connection with Commission Question 1, above. Thus the Board has a record before it which contains extensive information not previously available. In the course of the hearings comparative risk issues were addressed in testimony offered by Staff, licensees and intervenors from a number of different perspectives, including a comparison of safety levels at the Indian Point units with the Commission's preliminary safety goals, a comparison of the safety levels at Indian Point as determined by the IPPSS and Staff analyses with the results of site and plant specific PRAs conducted for a number of other nuclear power plants, an evaluation of the relevance of the Sandia Siting Study (NUREG/CR-2239), and a revisitation of the Task Force's 1980 comparative assessments in light of subsequent developments and a detailed engineering analysis of plant design features likely to be important to risk. We shall discuss each separately.

Indian Point Safety And The NRC Preliminary Safety Goals

The NRC published its Policy Statement on Safety Goals for the Operation of Nuclear Power Plants on March 14, 1983 (48 Fed. Reg. 10773). The Commission's preliminary safety goals are stated as quantitative design objectives which are expressed in terms of both individual and society risk. The preliminary safety goal Policy

Statement sets forth clear caveats that the goals are not to be used for licensing purposes, nor are they to be considered a de facto limit expressing levels of risk which should be deemed acceptable to the NRC. However, the preliminary safety goals do represent a national benchmark in terms of design objectives against which all nuclear power plants can be compared. Therefore, the comparison of numerical safety levels at the Indian Point plant as developed by the licensees and Staff under Commission Question 1 to the preliminary safety goals is one method of considering whether Indian Point is within the range of risks posed by other nuclear power plants.

In terms of individual risk, the Commission's preliminary safety goals state that the early fatality risk to an average individual in the vicinity (a one-mile radius) of a nuclear power plant should not exceed 1/10th of one percent of the sum of early fatality risk to that individual from other, non-nuclear "background" accidents. The United States national average accident risk is 5 fatal accidents per 10,000 persons per year (5×10^{-4} per year). (NUREG-0880 at p. 2) The early fatality risk from Indian Point to an individual within one mile of the plant is well within (i.e., lower than) the preliminary safety goal by a factor of approximately 70 for Indian Point Unit 2 and by a factor of approximately 75 for Indian Point Unit 3. Staff concurred that the early fatality risk at Indian

Point is below the preliminary safety goal, even when it employed the Staff's more conservative consequence analysis and combined the risk from both Indian Point units.

For societal risk, the Commission's preliminary safety goal specifies that the latent cancer fatility risk to the population in the vicinity (within 50 miles) of a nuclear power plant should not exceed 1/10th of one percent of the cancer fatality risks from other causes. The national average cancer risk is two fatalities per 1,000 persons per year (2×10^{-3} per year). (NUREG-0880 at p. 23) Comparing this average background latent cancer fatality risk within a 50-mile radius of the Indian Point facilities with the contribution to that risk from operation of the facilities as determined in IPPSS, the societal latent fatality risk from Indian Point is well within the safety goal by a factor of approximately 165 for Indian Point Unit 2 and by a factor of approximately 710 for Indian Point Unit 3. Based upon its conservative analysis, the Staff also agreed that the projected risk of latent cancer fatalities per year for the Indian Point site falls more than a factor of 10 below the preliminary safety goal.

Because of the large margins by which Indian Point meets the Commission's individual and societal risk preliminary safety goals, the significance of uncertainties in the respective licensee and Staff assessments is greatly

reduced. The use of more realistic source terms increases the margins between the assessed early fatality and latent cancer fatality risk and the preliminary safety goals still further. (Finding 5.0-11) Testimony by both licensees and Staff indicated that it is likely a significant reduction in severe reactor accident source terms may be forthcoming in the not too distant future. (Id.)

Testimony by the licensees and Staff also compared their respective calculations of core melt frequency with that of the preliminary safety goal, one in ten thousand (1×10^{-4}) per year of reactor operation. The licensees' assessment, which was based on current plant configurations, indicated median core melt frequencies from both internally and externally initiated events which are below the safety goal for Indian Point Unit 3 and which, when rounded off, are equal to the safety goal for Indian Point Unit 2. (Finding 5.0-12) Comparisons between the safety goal and median core melt frequencies for internal initiators only* indicate that both Indian Point units meet the goal. The Staff calculated core melt frequencies for both units at 3.5×10^{-4} per reactor year, which is slightly above the design objective. However, Staff's

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* Since the licensees' PRA is among the very few which have comprehensively addressed externally initiated events, we find that comparisons based on internal risk only are more appropriate at the present time. We address this subject more fully below.

analysis was based on acknowledged conservatisms, and did not give credit for certain subsequent plant modifications. (Finding 5.0-13)

In assessing the way in which Indian Point stacks up against the Commission's preliminary safety goals, the Board attaches much more significance to the individual (early fatality) and societal (latent cancer fatality) indices than to core melt frequency. We have accepted the principal finding of both the licensee and Staff risk assessments that core melt is a poor indicator of risk to public health and safety, since these analyses indicate that the great majority of core melt accidents at Indian Point would be contained by plant systems and would not lead to significant releases of radioactive material to the environment. The frequency of core melts accounting for nearly all of the offsite radiological risk was determined by both the licensees and Staff to be well below the quantitative safety goal of one such event in 10,000 years of reactor operation. (Finding 5.0-14) Even viewing the core melt frequency of the Indian Point units as falling above the safety goal does not suggest to us that they would be outside the range of risks posed by other nuclear power plants, since Staff has estimated that more than half of the plants in the United States may not meet the proposed safety goal for core melt frequency.

Comparisons Of Indian Point Risk With Other PRA Risk Results

One of the best ways to compare safety levels at one nuclear plant with those posed by other plants would be to compare site and plant specific PRAs for various plants, with each PRA being identical in scope and using state of the art methodology. However, the parties who submitted testimony on comparative safety agreed that the two dozen or so PRAs which had been published at the time of the hearings differed in scope, data base, methodology and sophistication. Some studies assessed only a few plant systems, while others did not address such things as externally initiated accidents. It was undisputed that only the IPPSS and Zion PRAs include a detailed treatment of the contribution of external events such as earthquakes, winds, fires and floods to the risk posed by those plants.

While these differences complicate the task of comparing safety between plants by comparing PRAs and limit our ability to make comparisons on such a basis, all of the published studies have analyzed the so-called internally initiated events in roughly the same manner. Therefore both the licensees and Staff utilized several PRAs to compare internally initiated event risk only. Internally initiated event risk for the Indian Point units was used to avoid an invalid comparison of Indian Point internal plus

external results with internal only results from other plants. Internally initiated events for which comparisons were made encompass spontaneous loss of coolant (LOCA) accidents, spontaneous transient events, LOCAs induced by spontaneous transients, and loss of offsite power. The licensees and Staff utilized PRAs done for the following plants to make such comparisons: Arkansas Nuclear One-1, Big Rock Point, Calvert Cliffs, Crystal River, Grand Gulf, Limerick, Oconee, Peach Bottom, Sequoyah, Surry, and Zion. The licensees made additional comparisons between Indian Point and the Millstone, Browns Ferry and Biblis (Germany) reactors. The PRAs selected for comparison generally comprise a representative sampling of plant designs, high and low population density sites, power levels, and reactor vendors and architect/ engineers.

Based on these comparisons, the Indian Point units are among the safer plants licensed to operate by the Commission with respect to the early and latent fatality risks to individuals posed by internally initiated serious reactor accidents. The testimony indicates that Indian Point is well below average in the frequency of severe releases caused by internally initiated accidents. This is confirmed not only by the bottom line PRA results, but by qualitative considerations and engineering judgments which we discuss below.

The population density surrounding a site has virtually no effect upon the risk to individuals. The Staff testified without contradiction that compliance with emergency preparedness requirements, and in particular the speed of an evacuation, also does not have an appreciable effect on individual risk. The situation is somewhat different when one looks to aggregate, societal risks, however. Societal risks are proportional to the population density distribution around a given site. Up to fifty miles, the population surrounding Indian Point is about ten times that of the median site, although other sites such as Zion and Limerick are quite similar. This puts the size of the population at risk at Indian Point (when WASH-1400 based source terms are used) near the top, but not well above the spectrum of population surrounding other plants licensed by the Commission. The various comparisons of the results of other PRAs to the results of the IPPSS and Staff analyses of the Indian Point plant for internal initiating events show that the early and latent fatality societal risk from Indian Point is within the range of risks estimated for other nuclear power plants. (Finding 5.0-44)

The licensees and Staff also made inter-plant comparisons of frequencies of severe releases of radioactivity from internally initiated events, based upon the published PRAs. A severe release is defined as a release which has the potential to cause offsite doses which

approach lethal levels. The frequency of severe releases is a much more relevant measure of risk to the public health and safety when one is trying to compare differences between plants. Comparisons of "internal" severe release probabilities as presented in the hearings demonstrate that the Indian Point units are among the best (i.e., have the lowest severe release frequencies) in comparison to the other nuclear power plants. (Finding 5.0-25) In the IPPSS analysis, the interfacing systems LOCA was found to dominate early fatality risk at Indian Point. This is, therefore, the major subset of severe releases at this plant; however, both the licensees and Staff examined the frequency of this event at Indian Point compared to that determined in other PRAs and concluded that the Indian Point units were less susceptible to this accident than other plants. (Findings 5.0-39 and 5.0-45)

For reasons which we have already discussed, the frequency of core melt is a poor indicator of risk at least for the Indian Point plant since the evidence demonstrates that the great majority of core melts would be contained without significant offsite releases of radioactivity. Nonetheless, based on the licensee and Staff comparisons of published PRAs, the internally initiated core melt frequencies for the Indian Point units are about average compared to other plants, and the uncertainty bounds for these frequencies are within the range of uncertainty bounds for other plants. (Finding 5.0-23)

The Implications Of The Sandia Siting Study In Assessing Comparative Plant Safety

During the course of the hearings a document entitled "Technical Guidance for Siting Criteria Development" (NUREG/CR-2239; SAND81-1549) was published, which has generally come to be known as the Sandia Siting Study. This document set forth certain accident consequence modeling for existing reactor sites which had been performed by Sandia National Laboratories under contract to the NRC for possible use in establishing future NRC reactor siting criteria. By our order of November 15, 1982 we asked the parties to address the significance of the Sandia Siting Study to Commission Question 5.

Using site-specific meteorology and population data and assuming an 1120 MWe reactor, the Siting Study assessed the consequences of hypothetical severe accidents at 91 nuclear plant sites throughout the United States. These consequences were modeled using an assumed "representative" set of generic Siting Source Terms (SSTs) which were intended to cover the full spectrum of releases for severe light water reactor accidents.

Since the entire purpose of the Siting Study was to evaluate site differences, the study arbitrarily assumed source terms and accident frequencies to be constant. Thus all parties including intervenors who commented upon the

Siting Study in their testimony acknowledged that no effort was made to model actual reactor/site combinations, that the uniform releases and frequencies assumed for purposes of the Study are not applicable to real plants, and that the Study results cannot be used to evaluate the safety levels at existing plants because this would also entail plant-specific probabilities of particular releases. The authors of the Siting Study in fact stated that "the results presented in this report do not represent nuclear power risk." (NUREG-2239 at p. iii) Thus the Siting Study presents a perspective on siting factors alone, and the consequence estimates represent only the consequences of three arbitrarily selected source terms which do not reflect any specific reactor design.

The Siting Study source terms are particularly at variance with those determined in the Staff and licensee analyses for Indian Point since the most frequently occurring release category at Indian Point, the late overpressurization release, has no Siting Study counterpart. In the Siting Study, the highest consequence SST-1 release was assumed to occur with about the same frequency as the Indian Point delayed overpressurization release, which results in a major overestimation of early fatality risk at Indian Point. (Finding 5.0-67) There were also warning time and protective response assumptions made in connection with Siting Study modeling which tended

to overstate consequences disproportionately at higher population density sites. (Finding 5.0-69) However, even given the Siting Study's modeling assumptions, when source terms other than the most serious SST-1 release were used in the Siting Study, they were found not to result in significant differences in public health and safety consequences between sites. The parties who addressed the Siting Study in their testimony agreed that when consequence calculations were performed in the Siting Study using other than the highest consequence source term, i.e. SST-2 and SST-3, variations between sites in terms of early fatalities, early injuries and latent cancer fatalities were found not to be significant. (Finding 5.0-59)

Intervenors contended that the Siting Study provides a basis for comparisons between sites when one assumes the occurrence of the Study's highest consequence SST-1 release. A comparison of the Indian Point consequence curves with those of other sites for the SST-1 release shows that the Indian Point curves fall into the upper range of curves for all sites in each of the three consequence indices used in the Study. The Staff testified that the Siting Study results assuming an SST-1 release and summary evacuation, show that Indian Point has higher mean early fatalities, early injuries, and latent cancer fatalities than the median site by factors of less than thirty, approximately fifteen, and approximately six,

respectively. However, given the very low absolute early fatality risks for Indian Point and for many other sites which were found in the Study, we are not prepared to conclude that variations of this magnitude are very important. A review if the Siting Study curves as aided by the testimony of the parties establishes that there are no apparent break points which would indicate that any one site or even a group of sites are out of the reasonable bounds of the family of curves. (Finding 5.0-61) Therefore, neither Indian Point nor any other site can be said to represent an extreme in terms of early fatality consequences for the Study's worst case release. The spread between sites is even smaller for the other two consequence indices. Thus the Siting Study site consequence modeling does not establish that the Indian Point site is unique or even unusual compared to the other 90 sites which were evaluated. It is clear, however, that when the Siting Study's most severe source term assumptions are used for modeling, the Indian Point site falls towards the high range of consequences but is not the highest.

Recent Indian Point Safety Assessments And The Task Force's Results

As we noted above, when the safety levels at Indian Point were first assessed by the Commission's Task Force in 1980 and then compared with several other

plant/site combinations, the Task Force's conclusion was that overall risk at Indian Point was about the same as a typical reactor on a typical site. However, compared with curves from the Task Force study, the early fatality societal risk curves for the Indian Point units as derived in IPPSS for internal initiators are lower by more than an order of magnitude, even though conservative source terms are used in both studies. A large part of this difference results from the fact that the Task Force did not evaluate the strength of the Indian Point containments, which have been found to preclude prompt failure. Thus the Task Force did not include a release category for late containment failure. The latent fatality data from IPPSS for internal initiators does not disturb and may even strengthen the Task Force finding. The IPPSS latent fatality risk curve lies within the range of the comparable Indian Point curves calculated by the Task Force, and is below the range of curves calculated for the typical pressurized water reactor (i.e., the Reactor Safety Study plant, Surry) at various sites.

The remaining element of the Task Force's basic conclusion which bears further scrutiny in light of the extensive plant evaluation undertaken in the IPPSS and Staff analyses relates to the special safety features of the Indian Point plant design. Since the Task Force found

the site demographics to increase societal risk,* but also found that the overall societal risk of Indian Point was about the same as a typical reactor on a typical site, this conclusion was necessarily dependent upon the Task Force's other finding that the Indian Point reactor design was about an order of magnitude "less risky than a typical design."

We noted earlier that the testimony supports the conclusion that Indian Point has a lower frequency of severe releases caused by internally initiated events, than many other plants, which is the PRA equivalent of saying, as did the Task Force, that the Indian Point reactor design is less risky. The record before us reveals several reasons why this is so. There are a number of engineered safety and other design features which have been

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* We should emphasize again that testimony in the record of this proceeding reaffirms the somewhat commonsensical observation that the density of population in the vicinity of a nuclear plant is virtually irrelevant to the risks posed to individuals, whose risk is a function of plant configuration and operation. If the Commission's safety objective was to equate risk to individuals from nuclear plants, the appropriate focus would be to bring plants with, for example, a greater frequency of serious releases up to some norm. In pursuit of such an objective it would be necessary to ignore for regulatory purposes differences in population between sites. The Board does not address whether this or some other objective may be required by the Atomic Energy Act or Constitutional principles.

incorporated into the Indian Point plant design to supplement standard safety features. Some of these special features were discussed at length in the Director's Orders of February 11, 1980. With the modeling of actual accident sequences in the event and fault trees of the Staff's and IPPSS analyses, the particular significance of these and other features has been identified.

Licensees testified that even using conservative source terms, the only accident found to contribute to early fatality risk at Indian Point was the interfacing systems LOCA. Staff testimony indicated that in addition to this sequence there were two others that had been generically identified as potential high risk sequences for pressurized water reactors, loss of all offsite power and failure of the auxiliary feedwater system, and special plant-specific vulnerabilities.

Staff stated that the Indian Point plant is less susceptible to the interfacing systems LOCA for two reasons. First, while most pressurized water reactors have three to six large pipes whereby such an accident could occur, Indian Point has but one. (Finding 5.0-30) Second, Indian Point has pressure boundary valves on that one line of the most reliable design. (Id.) Electric power is important to safety because most systems for cooling the core or removing heat from the containment are electrically powered. No other nuclear plant has as many sources of

backup power as Indian Point. With three emergency diesel generators for each unit and three gas turbine driven electrical generators Indian Point has a unique redundancy in backup sources of power. (Finding 5.0-31) Other plants have been found to have specific design vulnerabilities discovered through PRA even though all NRC licensing requirements are met. Yet at Indian Point, after one of the most thorough and intensive risk assessment scrutinies ever conducted, no such plant-specific deficiencies important to risk have been found for internally initiated accidents.

Among the other design features identified by the IPPSS and Staff analyses as important to risk are the containment building configuration, which allows gases to circulate easily and facilitates heat removal, the geometry of the reactor cavity and containment floor, which facilitates post-accident debris bed coolability, diverse containment cooling capability provided by five fan cooler units and four pumps capable of providing containment spray recirculation, two sumps for recirculation of emergency core cooling system water, confirmatory signals which reduce the likelihood of an inadvertent failure to isolate containment, a containment weld channel pressurization system to assist containment leak tightness, and a service water and component cooling water systems arrangement to maximize redundancy of components.

The Staff also compared the Indian Point units to other nuclear facilities with regard to their ability to prevent containment failure prior to core melt from a number of scenarios including overpressure failure, uncontained interfacing system loss of coolant accidents, earthquake-induced collapse of containment, internal and external missiles, open containment penetrations and steam generator tube ruptures with a stuck open atmospheric steam release valve. With regard to earthquake-induced failures of the containment, the testimony under Commission Question 1 has indicated that, based upon a reanalysis of the fragility of the Indian Point containments, such an earthquake-induced failure is not credible for the maximum conceivable earthquake at the site. (Finding 5.0-33) With regard to each of the other possible failure modes of the containment considered by the Staff, the Indian Point plants were found to be average or better than average compared to other nuclear plants in their ability to prevent these scenarios. (Id.)

The Staff also compared Indian Point to other nuclear plants with regard to the potential for containment failure at the time of core meltdown from pressure spikes and reactor vessel missiles, and found that the Indian Point units were better than average at containing pressure spikes associated with reactor vessel melt through, and no more susceptible to reactor vessel missiles than any other plant. (Finding 5.0-34)

Finally, the Staff evaluated the effectiveness with which an initially intact containment can confine core melt accidents both with an without the containment cooling and heat removal systems functional. The Indian Point containments appear to be among the more effective designs with regard to their capabilities to mitigate offsite radiological releases given a core melt. (Finding 5.0-35)

On the strength of the engineered safety features, containment design, and other Indian Point systems identified above, less than two percent of the internally initiated core melts were found to lead to containment failure at Indian Point. Staff concluded that Indian Point is well above average in its ability to withstand overpressure failure scenarios. (Id.)

The Indian Point containments, with a failure pressure limit of at least 141 psia and a volume of 2.6 x 10^6 cubic feet, are among the best in the country because of the large containment volume in proportion to reactor power level, the high failure pressure, and the use of basaltic concrete in the basemat, which would generate a lesser volume of gases in interaction with a molten core under the severe accident conditions.

The only other factor beyond the comparative likelihood of severe releases and the comparative effectiveness of emergency response which affects a comparison of individual risks is the weather for the

various sites. At the Indian Point site the weather is entirely typical of the distribution of weather conditions among the various reactor sites as they affect the consequences of accidents.

Indian Point Risk and External Events

Another major area in which risk insights have become available since the Task Force assessment three years ago is that of externally initiated events. Both individual and societal risks from external events are less clear because there is little basis for comparison among currently available PRAs, which generally do not address external accident risk. When externally initiated events were found in IPPSS to dominate Indian Point risk, three modifications were made to Unit 2 in order to reduce the contribution of external events. These modifications consisted of reducing the likelihood of interaction between Unit 2 and Unit 1 structures during severe earthquakes, alternate cable routing to assure operation of certain pumps in the event of a cable tunnel fire, and the establishment of anticipatory shutdown procedures should a hurricane threaten the site. After these changes the Staff concluded it is plausible that compared to other plants, Indian Point poses an average to below average risk to individuals from externally initiated events. Staff associated a great deal of uncertainty with this conclusion

due to the lack of comparable data on external event risk in most other PRAs. Yet Staff's conclusion is strengthened by other factors on which they did not rely, including alterations made to the control room ceilings at both units to reduce their seismic fragility, and perhaps more significantly, the reassessment of the ability of the two containments to withstand seismic events, which establishes that direct seismic failure would not occur. Staff also acknowledged that there were significant conservatisms in its hurricane analysis which tended to overstate risk from that cause.

There is also another source of assurance that the risk to individuals in the vicinity of Indian Point from externally initiated accidents is not larger than at other sites. The overall frequency of severe releases, which as we noted above is a key risk statistic from both internally and externally induced causes at Indian Point appears to be about average when compared to the frequency of severe releases from internal causes only at other plants. The frequency of severe releases could only go up at other plants if externally initiated events were included, yet Indian Point is roughly comparable even without adjusting the data for other plants.

Conclusion

The Board has heard evidence on the risk of Indian Point compared to other nuclear power plants using several different methods of comparison. Because of the absence of directly comparable risk assessments, it is not possible to state with quantitative precision how the safety levels of various plants compare. We do have before us a record which evaluates this issue from the standpoint of the Commission's preliminary safety goals, comparisons between PRAs, the Task Force's inquiry in 1980, and an evaluation of design features based on engineering judgment. Since the results of each approach guide us consistently in the same direction, we are more confident in reaching conclusions on comparative risk that we would be were just one basis of comparison available to us. The totality of the evidence is supportive of the conclusions reached in both the Staff's and licensees' testimony that the Indian Point containments are such that most serious core melt accidents would be contained. The containment, as well as other plant features at Indian Point which are not included at some other plants, causes the risk to individuals in the vicinty of Indian Point to be lower than average, which compensates for higher than average population and property values. Any comparisons among nuclear plants must take into account that there is a strong likelihood that

reductions in source term estimates would reduce differences in both individual and societal risk between plants, and could effectively eliminate the early fatality risk at all facilities.

COMMISSION QUESTION 6 AND CONTENTION 6.3

THE QUESTION AND CONTENTION PRESENTED

Question 6, as formulated by the Commission, reads:

"What would be the energy, environmental, economic or other consequences of a shutdown of Indian Point Unit 2 and/or Unit 3?".

Contention 6.3, which is really a part of Question 6, states:

"Considering the savings in operating expense which would result from shutting down Indian Point Units 2 and 3, and allowing for the ways in which cogeneration and conservation can mitigate the costs of replacement power, the net costs of shutdown are small; in fact, they are smaller than previous studies by UCS, GAO, or Rand suggest, and are entirely acceptable."

THE DIRECT ECONOMIC CONSEQUENCES OF A SHUTDOWN

THE ISSUES PRESENTED

All witnesses agreed that a shutdown of the plants would increase the Licensees' cost of supplying electric service in future years and would require commensurate increases in the rates charged by the Licensees to their customers in Con Edison's service area. However, the estimates of the increases in rates which the ratepayers would have to pay as the result of a shutdown differed

substantially, dependent primarily upon the witnesses' underlying assumptions as to the future prices of oil, the capacity factors projected for future operations of the plants, the projected operation and maintenance (O&M) expenses and capital charges associated with the Units, which would be avoided on a shutdown, the amount of energy which the Licensees' customers will take from them in the future, the number of years that the Units are likely to remain in operation, and the real discount rate used to determine the present value of future rate increases. Licensees' estimates also differed from the other estimates in that they were the only ones which showed the cost impact of a shutdown on a statewide basis as well as on the basis of Con Edison's service area. (Finding 6.0-3)

(a) The propriety of examining the cost impact on a statewide basis

The PROMOD III system used by the Licensees was the only program which provided an estimate of the direct cost impact of a shutdown in the State as a whole, as well as in Con Edison's service area. In Licensees' estimates the present value of future cost increases in New York State is $9.0 billion, which is $1.2 billion more than the $7.8-billion present value

for the cost increases projected for Con Edison's service Area. (Finding 6.0-5) Clearly, the total cost penalty, rather than a part of the cost penalty, is the more appropriate one to consider, and that alone is sufficient reason for using the PROMOD III System because it is the only program which provides a New York Power Pool dispatch necessary to estimate increased production costs on a statewide basis.

(b) The proper period to be covered by the estimate

In determining their projected net cost increases, Licensees used the expiration dates of the operating licenses of the Units as the end points of the periods to be examined. (Finding 6.0-7) Although some of the other parties used shorter periods, we do not see why the Units would not continue to operate until the expiration of their operating licenses and why shorter periods than those used by the Licensees would be appropriate.

(c) The best estimate of future oil prices

Ms. Streiter used forecasts of oil prices prepared by ICF, Inc. for the New York Power Pool in its 1983 submission to the State Energy Office. (Finding

6.0-7) The forecasts assumed that the current world price of $34 a barrel would decline 6.7 percent in 1983, return to the real 1982 price by 1985, and thereafter increase at a real rate of two percent per year. (Id.) Apart from ESRG, which tells us that the oil prices assumed in its mid-range case are somewhat higher than Licensees' oil prices (Finding 6.0-47(c) fn.), the other parties have all assumed that there will be no real increase in the price of oil over the long period they are talking about.

We adopt licensees' estimates of fuel prices for several reasons. First, ICF's 1985 forecasts are not out of line with the 1985 forecasts of other analysts referred to in Appendix 1 to Ms. Streiter's testimony, and its 1990 forecast is below the 1990 forecasts of those analysts. (Finding 6.0-45) Second, the conditions referred to by Mr. Conant, i.e., the use of more oil than is being discovered, the prospect of increased oil requirements by the less developed countries and by Communist nations, and the knowledge of every producer of oil that oil is a diminishing resource, support Mr. Conant's conclusion that a moderate economic recovery will strengthen oil prices. (Finding 6.0-44) Third, while some analysts may now be optimistic about the future, they were also optimistic

about the future in 1978, and the next year oil prices doubled. (Finding 6.0-45) Fourth, Licensees' estimates represent the views of an independent organization prepared in the normal course of business for use by members of the New York Power Pool in their long-range planning. (Finding 6.0-7) Finally, we see no reason to estimate oil prices on the low or "conservative" side, as Staff witnesses have done. (Finding 6.0-20) Our task, as we see it, is to present the most reasonable estimate.

(d) The best estimate of capacity factors

Ms. Streiter recognized that her use of a 63-percent capacity factor reflected a certain amount of optimism in light of the historical capacity factors of the Indian Point units. However, she pointed out that the lower capacity factors experienced in the past were the product of identified problems which have either been solved or are in the process of being solved, and in support of that, she noted that Con Edison achieved a capacity factor of 62 percent in its last refueling cycle, that the nuclear power industry has made great efforts to improve its response time to technical problems, and that the licensees have spent and will

spend huge amounts to improve the operations of the Indian Point Units. (Finding 6.0-11)

While Ms. Streiter thus made a persuasive case for 63 percent, she recognized that one could defend the use of 57 percent in the Licensees' reference case. (Finding 6.0-13) In that connection, she testified that the use of a 57-percent capacity factor would reduce her present value of the ongoing future costs of a shutdown from $9.0 billion to $7.583 billion. (Id.)

Based on this testimony, we have found that the average capacity factor of the Indian Point units would most likely vary between 57 and 63 percent. In fixing this range, we have considered and rejected claims that the capacity factor should be lower than 57 percent.

Staff has recommended a capacity factor of 50 percent in order to be "conservative." (Finding 6.0-12) We do not see why Staff should recommend, or we should find, a "conservative" capacity factor, as distinguished from what we think is the most likely capacity factor. And 50 percent does not appear to be Staff's estimate of the most likely capacity factor. (Id.)

Messrs. Schlissel and Mavretich think that the historical capacity factors of the Indian Point Units are the best numbers that we have. (Finding 6.0-27) However, Ms. Streiter has argued persuasively that the

past is not likely to be prologue in this instance, and that the Licensees have solved or are in the process of solving the problems responsible for their low capacity factors in the past. (Finding 6.0-11)

Dr. Rosen purported to find support for ESRG's use of an average capacity factor of approximately 43 percent in testimony given by Dr. Louis J. Perl of NERA and in various regression analyses made by ESRG. (Finding 6.0-30) Neither of these references supports the use of these very low capacity factors.

Dr. Perl's generic estimate was that capacity factors would be approximately 50 percent at the outset of a unit's operations, increase to around 70 percent at the end of ten years of operation, remain at 70 percent for about 10 years, and then fall steadily to 20 percent at the end of the unit's book life. (Finding 6.0-30) When Dr. Rosen was asked why he was willing to accept Dr. Perl's end point, but not his intermediate points, his answer was that there was no operating data to support Dr. Perl's intermediate points. (Id.) Needless to say, there were less data -- in point of fact, no data -- for Dr. Perl's end points.

The regression analyses relied on by Dr. Rosen are also unconvincing because no unit which is the size of the Indian Point units has been operating for more

than ten years. (Streiter Testimony, p. 9; Streiter, Tr. 13492-96; Rosen, Tr. 13853) In light of the absence of an adequate data base for ESRG's regression analysis, it is not surprising to find that the actual 1982 capacity factors for saltwater PWR's were significantly larger than the capacity factors which that regression analysis predicted for them for 1982. (Buehring, et al., Tr. 14131-33)

Ms. Streiter has also convincingly demonstrated that the so-called "age effect", which Dr. Rosen purported to find in nuclear plants between "years six or seven to perhaps ten or twelve" (Finding 6.0-30), is almost entirely due to recent steam generator problems at three units, and the data with respect to one of those three units alone are statistically responsible for half of the so-called "age effect" -- although the extended outage of that unit involved seismic related improvements, and had nothing to do with saltwater. (Id.)

(e) The most reasonable estimate of O&M expenses

To arrive at the O&M offset to the production cost increases attributable to a shutdown, Ms. Streiter, used Licensees' detailed estimates to 1986 and assumed

that after 1986 O&M expenses will increase at a real rate of one percent per year. (Finding 6.0-7) While those estimates were considerably higher than the United States average per kw (Finding 6.0-47(e)), they were considerably lower than the O&M cost projected by ESRG. (Finding 6.0-31)

To arrive at its projection of O&M costs, ESRG applied a simple linear least squares fit to Licensees' experienced costs. However, ESRG made no attempt to determine the nature of the recent increases in O&M costs and the propriety of extrapolating those increases on a linear basis. (Id.) Ms. Streiter made such an analysis, and her analysis showed that these increases could not properly be extrapolated into future years because a large part of the recent increases consists of costs which will continue but not grow in the future and of a series of one-time expenditures related to new regulatory requirements or specific operating problems. (Finding 6.0-32) In light of that analysis, we reject ESRC's estimate and adopt the licensees' estimate of O&M costs.

(f) The most reasonable estimate of future energy demands

There are basically three forecasts of energy consumption in the record. The State Energy Office has used a statewide growth rate of 1.7 percent, which is the rate used in the official New York State Energy Master Plan (SEMP II) forecast (Finding 6.0-47(d)); the Licensees have used a statewide growth rate of 1.4 percent, and a growth rate of 1.3 percent in Con Edison's service area (Finding 6.0-7); and ESRG has used a zero (or slightly negative) growth between 1983 and 1997. (Finding 6.0-33)*

We reject ESRG's forecast of energy consumption as unreasonably low on several grounds.

First, the ESRG Report assumes a 4.2-percent reduction in energy consumption in Con Edison's service area in the five-year period 1983-1988 (Finding 6.0-33), which is inconsistent with the 4.1 percent growth experienced in that area between 1977 and 1982. (Id.)

Second, ESRG states that its demand growth scenarios were based upon its June 1981 study for the

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* As is the case with respect to capacity factor, the negative growth cannot be found in the direct text of ESRG's Report.

New York City Energy Office (Id.) In point of fact, however, the 1981 study showed a growth rate of 0.5 percent for its mid-range case. (Id.)

Third, ESRC purports to justify its zero or negative growth assumption by reference to a conservation scenario included in its June 1981 study. (Id.) Apart from the fact that that conservation scenario is not included in the record before us and does not become part of the record simply because it is referenced in a footnote, evidence that is in the record persuades us that:

a) The Licensees and the State Energy Office took adequate account of the potential for conservation (including conservation achievable through cost-justified subsidies) in their forecasts of growth in energy consumption (Finding 6.0-37); and

b) Given the low price elasticity of demand in the Con Edison service area (Finding 6.0-43), it is unlikely that the closing of the Indian Point Units would spur an increase in conservation. (Finding 6.0-22)

In light of our conclusions with respect to ESRC's estimate of the growth in energy consumption, our choice is between the 1.4 percent advanced by the Licensees and the 1.7 percent advanced by the State Energy Office. Here again we are constrained to adopt a

range -- 1.4 percent to 1.7 percent -- rather than a single rate of growth. From Ms. Streiter's testimony that the use of her high load growth of 2.1 percent would produce a present value of $10.85 billion, and that the use of her low load growth of 0.7 percent would produce a present value of $7.49 billion (Finding 6.0-14), we can tell that a difference of 1.4 percentage points in growth accounts for a difference of $3.36 billion in present value. This, in turn, means that a difference of one percentage point in growth accounts for a difference of $2.4 billion in present value, and that a difference of 0.3 of a percentage in growth accounts for a difference of $720 million in present value.

(g) The most appropriate real discount rate

The real discount rates used to determine the present value of a stream of future rate increases ranged from Licensees' three percent to Staff's five percent. In the absence of any discussion as to why one rate is more appropriate than another, we adopt a real discount rate of three percent because three percent has for many years been referred to as the real rate of interest. See, e.g., Beryl Wayne Sprinkel, Money and

Markets -- A Monetarist View (1971), Ch. 8, pp. 206-16; Frank K. Reilly, Investments (1982), pp. 12-19.

THE BEST ESTIMATE OF THE DIRECT ECONOMIC COST OF A SHUTDOWN

Based on our resolution of the aforesaid issues, we conclude that a permanent shutdown of the Indian Point Units at this time would result in substantial future rate increases to the ratepayers in New York State, and most particularly to the ratepayers in Con Edison's service area, and that the present value of these ongoing rate increases is likely to range between $7.6 billion and $9.7 billion. The low end of our range reflects the use of a capacity factor of 57 percent. The upper end of our range reflects an annual growth rate in energy consumption of 1.7 percent.

We recognize, of course, that our estimate is heavily judgmental because of the uncertainties inherent in the choice of assumptions which underlie and affect any estimate of the net cost of a shutdown. We believe, however, that any reasonable estimate would mandate the conclusion that a shutdown would impose a substantial economic burden on the ratepayers in the Con Edison service area.

In reaching that conclusion we have considered and rejected claims that the economic impact of a shutdown could be offset by a conservation program that induces the accelerated replacement of energy inefficient appliances. (Finding 6.0-35) We find that the proponents of these claims did not really consider where the billions of dollars necessary to purchase the new appliances would come from or what the annual cost to the Licensees' ratepayers would be if they had to pay a return on the capital needed to finance those purchases. (Finding 6.0-36) Moreover, we find that there is no evidence that the closing of Indian Point would significantly accelerate the adoption of such conservation options as are already available. On the contrary, there is evidence that this would not happen. (Finding 6.0-22)

Finally, we do not understand how the expenditure by the Licensees of billions of dollars to replace appliances, which their owners do not apparently consider are cost-effective to replace, would in any way offset the costs of closing the plant. Indeed, it would appear to add to the costs.

We have also considered and rejected contentions that the economic impact of a shutdown could be substantially mitigated by cogeneration. As Staff

properly recognized, cogeneration could only mitigate the impact of a shutdown if the rates for the power that Licensees buy back from the cogenerators were at a fixed price that was less than the Licensees' avoided costs. (Finding 6.0-22) Given the Supreme Court's recent affirmance of FERC's holding that PURPA requires a buy-back rate equal to avoided costs (American Paper Institute Inc. v. American Electric Power Service Corp., 51 U.S. Law Week 4547, 4549-51 (May 16, 1983)), the possibility of a mitigation of the cost impact of a shutdown through cogeneration has disappeared. Moreover, the spread of cogeneration would be limited by physical constraints, such as air pollution problems and natural gas supply problems. (Findings 6.0-47(i), and 6.0-38 through 6.0-42)

THE INDIRECT ECONOMIC CONSEQUENCES OF A SHUTDOWN

In addition to the direct economic burden which a shutdown would impose upon Licensees' ratepayers, there will in our opinion be two indirect economic costs. The first of these is that the MTA will be hit particularly hard by the rate increases which would follow a shutdown. Whether those rate increases would be covered by the MTA through its own rate increase or through a

cut in the quality of its service, the end result could be a loss in jobs, a loss in personal income, and a loss in tax revenues in New York City. (Finding 6.0-57)

The second indirect economic impact involves the possibility that the Power Authority, because of competitive conditions, would not be able to pass all its required rate increases on to its customers in the Con Edison service area, but would have to impose them on its hydroelectric customers in Upstate New York. (Finding 6.0-58) Were that to take place, those who pick up the tab for the shutdown would not be the people who would purportedly benefit from the diminishment of risk attributable to a shutdown.

THE ENVIRONMENTAL AND REGIONAL SOCIOECONOMIC CONSEQUENCES OF A SHUTDOWN

A shutdown would not have any adverse environmental consequences. (Finding 6.0-62) A shutdown would, however, adversely affect the local areas which depend substantially upon taxes attributable to the operation of the Indian Point Units, and the resultant increase in other taxes or the reduction in services in those areas would have an especially adverse impact on those living on fixed incomes such as the elderly. (Findings 6.0-61 and 6.0-63)

CONTENTION 6.1

WBCA has presented the contention that Rockland County would benefit economically from a shutdown because Orange and Rockland has excess capacity and would sell replacement power. That contention must be rejected because Orange and Rockland is a net purchaser of energy, notwithstanding its excess capacity, and would suffer an increase in its production costs as the result of a shutdown. (Finding 6.1-4)

CONTENTION 6.2

Parents contend that a shutdown would be beneficial because the environment of children in the vicinity of the Indian Point Units would be improved by a decrease in the release of radioactive material. However, the undisputed testimony is that the risk to a child exposed to radioactive effluents from one reactor-year of normal operation is a small fraction of the risk from one year of exposure to natural background radiation; that the potential of genetic disorders from such radioactive effluents is a very small part of the estimated normal incidence of genetic abnormalities; and that Parents' own witnesses agreed that the contribution of Indian

Point's routine operations is absolutely miniscule. (Findings 6.2-1 through 6.2-5) Given that testimony, we must reject Contention 6.2 because we could not find anything beyond an absolutely insignificant or de minimus improvement in the environment -- if we could, in fact, find that.

BOARD RECOMMENDATIONS

Recommendation No. 1

The licensees should continue their ongoing program of utilizing probabilistic risk assessment methodology to identify and examine potentially cost effective plant or procedural modifications which significantly reduce risk. In addition, the licensees should perform revised IPPSS consequence model risk calculations after safety goals for operating plants have been determined and ongoing degraded core and source term research completed.

Discussion

The licensees should continue to utilize the IPPSS to provide additional risk insights into the significance of licensing changes or plant modifications. While the results to be derived from a PRA such as the IPPSS cannot be directly used as the primary justification either for or against a particular change or modification, they can be used to put into perspective their relative significance. The Board feels this is the most beneficial use of a PRA over the next several years while the implications of preliminary safety goals to operating plants are still being determined and while additional research and rulemaking in the areas of degraded core phenomenology, siting considerations and source term reduction are underway. Since significant changes to PRA input parameters and assumptions may result from this ongoing

work, we believe that it would be most beneficial, both technically and economically, to defer rerunning or maintaining the currency of the IPPSS until such time as the rulemaking and research issues are resolved and safety goals are firmed. Since the Board finds that the risks posed by the Indian Point plants are within the range of risks posed by other nuclear facilities, we find no immediate need for the Indian Point licensees to take any further action in the area of PRA not being required generically at this time.

Recommendation No. 2

Rockland County should rejoin the radiological emergency planning process on a coordinated basis with the State of New York and the other three counties in the vicinity of Indian Point.

Discussion

Neither Rockland County nor its citizens have been well served by the decision of its political leadership to "go it alone" in preparing for the possibility of a serious radiological accident at Indian Point which could affect Rockland County citizens. The diligence with which Rockland County has been moving forward in preparing its own plan has not matched the apparent perception of the county leaders as to the dangers posed by the plant which have motivated their concerted efforts to bring about a

shut down. With the willingness of the State of New York to compensate for the lack of Rockland County participation through the use of State resources, any opportunity which Rockland County had to shut the plant down through a lack of participation in emergency planning has not proven productive. Since Indian Point is presently in operation, the best interests of the citizens of Rockland County would be furthered by a decision by the county leadership to rejoin the coordinated emergency planning process from which they withdrew in May 1982. By so doing, the County would obtain benefits from interaction with the State and other governmental entities which are presently unavailable to it.

Recommendation No. 3

FEMA should be encouraged to develop a program for field training of prospective radiological emergency planning workers, with such training given in the vicinity of the affected site.

Discussion

While FEMA offers comprehensive educational and training programs in techniques important to radiological emergency planning, such training is presently provided primarily at locations in Maryland and Nevada. This circumstance diminishes the access which prospective radiological emergency planning workers have to FEMA's

resources. Were FEMA to develop a program of field training, with courses to be offered on a scheduled basis in the general vicinity of nuclear plant sites, a much greater number of prospective workers would be able to receive such training.

Recommendation No. 4

FEMA should be encouraged to develop a model training manual containing general radiological information for use in training emergency workers.

Discussion

The record before us indicates that printed training materials used on a state-wide basis in the State of New York contain some erroneous information. We would be surprised if this were not the case elsewhere, with each state most likely attempting to develop written training materials almost on an ad hoc basis. Since the required radiological information is generic, it would be advantageous if model training materials were developed by FEMA for national use in training emergency workers.

Recommendation No. 5

NRC and FEMA should revise the emergency planning guidance document, NUREG-0654, to require consideration of plant-specific information.

Discussion

While NUREG-0654 presently requires extensive consideration of the particular characteristics of the area surrounding a nuclear power plant in connection with emergency planning, the document does not at present require that parallel consideration be given to the characteristics of the plant which would cause the accident being planned for. At Indian Point, the uncontroverted evidence shows that the great majority of serious accidents which might be expected would be of the delayed (or so-called "late overpressurization") variety. This is very useful information for purposes of emergency planning, yet it is not considered under the present NRC and FEMA guidelines. Thus, for example, FEMA emergency planners at Indian Point rejected emergency planning measures involving personnel who could not be available in the vicinity of the plant in less than four or five hours notice. Presumably, this would not have occurred had plant-specific considerations been factored into the decisional process. For plants with comprehensive probabilistic risk assessments, plant-specific characteristics should be given consideration equivalent to site-specific characteristics in the emergency planning process.

Recommendation No. 6

NRC should review justification for the 10-mile plume exposure pathway Emergency Planning Zone (EPZ).

Discussion

The about 10-mile radius required for comprehensive radiological emergency planning under the Commission's current regulations was derived from a 1978 analysis published as NUREG-0396. Much information about radionuclide behavior and accident modeling has become available since that time. Testimony given before the Board by NRC Staff witnesses indicates that the source terms used in the NUREG-0396 analysis were quite conservative and that an understanding of the conservatisms inherent in using such source terms is essential for informed regulatory decisionmaking. Because of the major significance associated with an NRC requirement of comprehensive emergency planning within about a 10-mile radius of each site, the appropriateness of the radius to the accidents being planned for should be updated.

Recommendation No. 7

FEMA and NRC should agree on guidelines for regional offices characterizing emergency planning deficiencies as "significant" or "minor" based upon their probable impacts on public health and safety.

Discussion

The record before us indicates that in connection with several evaluations of radiological emergency planning at Indian Point in 1982 and 1983, FEMA evaluators characterized emergency planning deficiences as "significant" or "minor" without receiving any guidance from the NRC or from FEMA's national office. In making such characterizations themselves, the record indicates that the FEMA evaluators did not base their determinations of significance upon the probable impacts on public health and safety of the particular deficiency. Given the major importance which the current regulatory scheme attaches to whether or not a given emergency planning deficiency is characterized as "significant" or in some other way, much greater attention should be given to the manner in which significance is assessed. NRC has been able to develop a clear set of guidelines for assessing the significance of licensing violations through the Severity Levels of its enforcement program. These Severity Levels require that distinctions be drawn between the important and less important based upon specific criteria. No less should be tolerated in the area of radiological emergency planning.

Recommendation No. 8

The 1984 edition of the Indian Point radiological emergency planning booklet should be revised to better reflect the early school dismissal plan.

Discussion

In the course of the March 9, 1983 Indian Point emergency planning exercise, an option was selected in each county whereby schools were dismissed under generic state law "snow day" procedures at the alert stage of the scenario. The record before us indicates that early school dismissal would now be looked upon as an attractive option in the event of an actual emergency at Indian Point. This represents a departure from prior practice. Accordingly, the next edition of the Indian Point emergency planning brochure should be revised to emphasize the early school dismissal option to a greater extent.

Recommendation No. 9

The State of New York should encourage the enhancement of the Emergency Broadcast System (EBS) to include ribbon television announcements in the event of an emergency as a supplement to the present radio announcements.

Discussion

An intervenor witness suggested that communication by television through a so-called ribbon announcement at the top or bottom of the television screen would reach a broader audience, and more quickly, in the event of an actual emergency than would current reliance upon radio announcements alone. The Board believes that this

suggestion has merit to the extent that ribbon announcements are not presently provided for on all stations serving the Indian Point plume exposure EPZ, and should be explored by the State of New York. Televised EBS messages would also help overcome problems associated with the needs of deaf persons in the event of a radiological emergency, as referred to by several witnesses.

ATTACHMENT A

LIST OF WITNESSES - QUESTION 1

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Acharya Testimony	Testimony of Dr. Sarbeswar Acharya Regarding NRC Staff Assessment of Accident Consequences and Risks	Sarbeswar Acharya	Senior Radiological Engineer in the Radiological Analysis Section of the Accident Evaluation Branch, Division of Systems Integration Within the Office of Nuclear Reactor Regulation of the NRC	NRC Staff	ff. Tr. 8566
Acharya/ Blond Testimony	Joint Testimony of Sarbeswar Acharya and Roger M. Blond Regarding Calculational Methodology of the CRAC Code	Sarbeswar Acharya	Senior Radiological Engineer in the Radiological Analysis Section of the Accident Evaluation Branch, Division of Systems Integration Within the Office of Nuclear Reactor Regulation of the NRC	NRC Staff	ff. Tr. 8568
		Roger M. Blond	Section Leader of the Accident Risk Section, Reactor Risk Branch, Division of Risk Analysis, Office of Research - NRC	NRC Staff	ff. Tr. 8568

LIST OF WITNESSES - QUESTION 1

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Bernero Testimony	Testimony of Robert M. Bernero on Severe Accident Source Terms	Robert M. Bernero	Director of the Accident Source Term Program Office in the Office of Nuclear Regulatory Research at the NRC	NRC Staff	ff. Tr. 12,581
Blond 1.3	Director Testimony of Roger Blond Concerning Board Question 1.3	Roger Blond	Section Leader for the Accident Risk Section of the Reactor Risk Branch of the Division of Risk Analysis of the Office of Research	NRC Staff	ff. Tr. 8369
Buchbinder/ Kubicki Testimony	Direct Testimony of Benjamin Buchbinder and Dennis Kubicki Concerning Commission Question 1	Benjamin Buchbinder	Chief, Transportation and Materials Risk Branch, Division of Risk Analysis, Office of Nuclear Regulatory Research - NRC	NRC Staff	ff. Tr. 7577
		Dennis Kubicki	Fire Protection Engineer in the Chemical Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation - NRC	NRC Staff	ff. Tr. 7577
Budnitz Testimony	Direct Testimony of Robert J. Budnitz Concerning Commission Question 1	Robert J. Budnitz	President of Future Resources Associates	NRC Staff	ff. Tr. 7489

LIST OF WITNESSES - QUESTION

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Codell Testimony	Direct Testimony of Richard Codell Concerning Commission Question 1	Richard Codell	Senior Hydraulic Engineer in the Hydrologic and Geotechnical Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation - NRC	NRC Staff	ff. Tr. 8578
Geiger/ Sidel Testimony	Direct Testimony of Dr. H. Jack Geiger and Dr. Victor W. Sidel On Behalf of Friends of the Earth, Inc. and the New York City Audubon Society	H. Jack Geiger	Arthur C. Logan Professor of Community Health at the City College of the City University of New York and Chairman of the Department of Community Health and Social Medicine in the School of Biomedical Education at City College	Friends, Audubon Society	ff. Tr. 8481
		Victor W. Sidel	Chairperson of the Department of Social Medicine at Montefiore Medical Center and Professor of Community Health at the Albert Einstein College of Medicine	Friends, Audubon Society	ff. Tr.8481

LIST OF WITNESSES - QUESTION 1

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Israel, et al.	Direct Testimony of Sanford Israel, Jack Hickman, Gregory Kolb, Robert G. Easterling, and Alan D. Swain, Commission Question 1 and Board Question 1.1	Sanford Israel	Risk Analyst in the Reliability and Risk Assessment Branch of the Division of Safety Technology Within the Office of Nuclear Reactor Regulation of the NRC	NRC Staff	ff. Tr. 7391
		Jack Hickman	Supervisor of the Nuclear Fuel Cycle Systems Safety Division of Sandia National Laboratories	NRC Staff	ff. Tr. 7391
		Gregory Kolb	Member of the Technical Staff Within the Nuclear Fuel Cycle Systems Safety Division at Sandia National Laboratories	NRC Staff	ff. Tr. 7391
		Robert G. Easterling	Staff Member in the Reliability Department of Sandia National Laboratories	NRC Staff	ff. Tr. 7391
		Alan D. Swain	Member of the Technical Staff in the Statistics, Computing and Human Factors Division at Sandia National Laboratories	NRC Staff	ff. Tr. 7391

LIST OF WITNESSES - QUESTION 1

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Joyce Testimony	Direct Testimony of Joseph P. Joyce Concerning Commission Question 1, Contention 1.1	Joseph P. Joyce	Senior Engineer in the Instrumentation and Control Systems Branch of the Division of Systems Integration	NRC Staff	ff. Tr. 8692
Kaplan 1.2	Licensees' Testimony of Stanley Kaplan on Board Question 1.2	Stanley Kaplan	Associate Consultant at Pickard, Lowe and Garrick, Inc.	Con Edison/ PASNY	ff. Tr. 8244
Levi Testimony	Direct Testimony of Isaac Levi, Ph.D. on Behalf of Friends of the Earth, Inc. and New York City Audubon Society	Isaac Levi	Professor of Philosophy at Columbia University	Friends, Audubon Society	ff. Tr. 7716
Licensees 1	Licensees' Testimony on Commission Question One and Board Question 1.1	John Garrick	Principal, Pickard, Lowe and Garrick, Inc., Specialist in Nuclear Safety Matters	Con Edison/ PASNY	ff. Tr. 6961
		David K. Goeser	Manager of Marketing in the Nuclear Technology Division of Westinghouse Electric Corporation	Con Edison/ PASNY	ff. Tr. 6961

LIST OF WITNESSES - QUESTION 1

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Licensees 1 (Continued)	Licensees' Testimony on Commission Question One and Board Question 1.1	Stanley Kaplan	Associate Consultant at Pickard, Lowe and Garrick, Inc. in Probability Theory, Risk and Decision Analysis	Con Edison/ PASNY	ff. Tr. 6961
		Dennis C. Bley	Consultant at Pickard, Lowe and Garrick, Inc. in Reliability, Risk, and Decision Analysis for Electrical Generating Plants	Con Edison/ PASNY	ff. Tr. 6961
		Robert E. Henry	Vice President and Co-Founder of Fauske & Associates, Inc.	Con Edison/ PASNY	ff. Tr. 6961
		Nicholas J. Liparulo	Core, Containment, and Sequence Analysis Group Manager of the Risk Assessment Technology Group in the Nuclear Safety Department of the Nuclear Technology Division of Westinghouse Electric Corporation	Con Edison/ PASNY	ff. Tr. 6961

LIST OF WITNESSES - QUESTION 1

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Licensees 1 (Continued)	Licensees' Testimony on Commission Question One and Board Question 1.1	Harold F. Perla	Structural Engineer and Associate Consultant at Pickard, Lowe and Garrick, Inc.	Con Edison/ PASNY	ff. Tr. 6961
		Dennis C. Richardson	Risk Assessment Technology Manager in the Nuclear Safety Department of the Nuclear Technology Division of Westinghouse Electric Corporation	Con Edison/ PASNY	ff. Tr. 6961
		Richard H. Toland	Manager of the Structural Analysis Group of United Engineers and Constructors, Inc.	Con Edison/ PASNY	ff. Tr. 6961
		Thomas E. Potter	Consultant at Pickard, Lowe and Garrick, Inc., in Public Health Consequence Analysis of Radioactive Releases	Con Edison/ PASNY	ff. Tr. 6961
		Dee H. Walker	Manager of Nuclear Engineering in the Nuclear Safety Department of the Nuclear Technology Division of Westinghouse Electric Corporation	Con Edison/ PASNY	ff. Tr. 6961

LIST OF WITNESSES - QUESTION 1

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
T. Meyer 1.4	Power Authority's Testimony of Theodore A. Meyer on Board Question 1.4	Theodore A. Meyer	Manager of the Division of Reactor Vessel Integrity Group of the Nuclear Technology Division of Westinghouse Electric Corporation	Power Authority	ff. Tr. 13,091
Meyer/Pratt Testimony	Direct Testimony of James F. Meyer and W. Trevor Pratt Concerning Commission Question 1	James F. Meyer	Senior Task Manager - NRC	NRC Staff	ff. Tr. 12,492
		W. Trevor Pratt	Group Leader for the Accident Evaluation Group in the Division of Engineering and Risk Assessment at Brookhaven National Laboratory	NRC Staff	ff. Tr. 12,492
Nesse Testimony	Direct Testimony of Ronald J. Nesse On Site-Specific Economic Input Parameters	Ronald J. Nesse	Research Economist With the Energy Systems Department of Pacific Northwest Laboratory	NRC Staff	ff. Tr. 8575
Palenik/Beyea Testimony	Preliminary Testimony of June 7, 1982, "Some Consequences of Catastrophic Accidents at Indian Point and Their Implications for Emergency Planning", Direct Testimony of Brian Palenik and Dr. Jan Beyea	Brian Palenik	Senior Energy Scientist for the National Audubon Society	UCS/NYPIRG, Audubon Society, New York State Attorney General	ff. Tr. 2900
		Jan Beyea	Staff Member of National Audubon's Policy Research Department		

LIST OF WITNESSES - QUESTION 1

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Palenik/Beyea Supp. Testimony	Supplements to Preliminary Testimony of June 7, 1982, "Some Consequences of Catastrophic Accidents at Indian Point and Their Implications for Emergency Planning", Direct Testimony of Brian Palenik and Dr. Jan Beyea	Brian Palenik	Senior Energy Scientist for the National Audubon Society	UCS/NYPIRG, Audubon Society, New York State Attorney General	ff. Tr. 2900
		Jan Beyea	Staff Member of National Audubon's Policy Research Department		
Perrow Testimony	Direct Testimony of Charles Perrow, Ph.D. on Behalf of Friends of the Earth, Inc. and the New York City Audubon Society on Commission Question 1	Charles Perrow	Tenured Professor of Sociology at Yale University	Friends, Audubon Society	ff. Tr. 7843
Pisello/ Piccioni Testimony	Direct Testimony of Daniel M. Pisello, Ph.D. and Richard G. Piccioni Ph.D. on Behalf of Friends of the Earth and the New York City Audubon Society	Daniel M. Pisello	Assistant Professor of Physics at the New York Institute of Technology	Friends, Audubon Society	ff. Tr. 8086
		Richard G. Piccioni	Assistant Professor in the Department of Biological Sciences at Hunter College of the City University of New York	Friends, Audubon Society	ff. Tr. 8086

LIST OF WITNESSES - QUESTION 1

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Potter 1.3	Licensees' Testimony of Thomas E. Potter on Board Question 1.3	Thomas E. Potter	Consultant at Pickard, Lowe and Garrick, Inc. in Public Health Consequence Analysis of Radioactive Releases	Con Edison/ PASNY	ff. Tr. 8346
Reed Testimony	Direct Testimony of John W. Reed Concerning Commission Question 1	John W. Reed	President of Jack R. Benjamin and Associates, Inc. (JBA)	NRC Staff	ff. Tr. 7492
Richardson/ Bley 1.4	Licensees Testimony of Dennis C. Richardson and Dennis C. Bley on Board Question 1.4	Dennis C. Richardson	Risk Assessment Technology Manager in the Nuclear Technology Division of Westinghouse Electric Corporation	Con Edison/ PASNY	ff. Tr. 8382
		Dennis C. Bley	Consultant at Pickard, Lowe and Garrick, Inc., in Reliability, Risk, and Decision Analysis for Electric Generating Plants	Con Edison/ PASNY	ff. Tr. 8382
Rowsome 1.2	Direct Testimony of Frank H. Rowsome Concerning Board Question 1.2	Frank H. Rowsome	Deputy Director of the Division of Risk Analysis Within the Office of Nuclear Regulatory Research - NRC	NRC Staff	ff. Tr. 8277

LIST OF WITNESSES - QUESTION 1

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Rowsome III.A.4	Direct Testimony of Frank H. Rowsome Concerning Damage State Likelihoods	Frank H. Rowsome	Deputy Director of the Division of Risk Analysis in the Office of Nuclear Regulatory Research - NRC	NRC Staff	ff. Tr. 7597
Rowsome IV.C	Direct Testimony of Frank H. Rowsome Concerning IV.C Accuracy of the Risk Assessments	Frank H. Rowsome	Deputy Director of the Division of Risk Analysis in the Office of Nuclear Regulatory Research - NRC	NRC Staff	ff. Tr. 8778
Rowsome V	Direct Testimony of Frank H. Rowsome to Contention 1.1 and Board Question 1.1	Frank H. Rowsome	Deputy Director of the Division of Risk Analysis in the Office of Nuclear Regulatory Research - NRC	NRC Staff	ff. Tr. 8780
Rowsome/ Blond I	Direct Testimony of Frank H. Rowsome and Roger Blond Concerning Commission Question 1	Frank H. Rowsome	Deputy Director of the Division of Risk Analysis in the Office of Nuclear Regulatory Research - NRC	NRC Staff	ff. Tr. 7169
		Roger Blond	Section Leader for the Accident Risk Section of the Reactor Risk Branch of the Division of Risk Analysis of the Office of Research - NRC	NRC Staff	ff. Tr. 7169

LIST OF WITNESSES - QUESTION 1

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Rowsome/ Blond IV.A	Direct Testimony of Roger M. Blond and Frank H. Rowsome, Summary Response to Commission Question Posed to the Board	Roger M. Blond	Section Leader for the Accident Risk Section of the Reactor Risk Branch of the Division of Risk Analysis of the Office of Research - NRC	NRC Staff	ff. Tr. 8771
		Frank H. Rowsome	Deputy Director of the Division of Risk Analysis in the Office of Nuclear Reactor Research - NRC	NRC Staff	ff. Tr. 8771
Rowsome/ Blond IV.B	Direct Testimony of Frank Rowsome and Roger Blond Concerning Commission Question 1	Frank Rowsome	Deputy Director of the Division of Risk Analysis in the Office of Nuclear Reactor Research - NRC	NRC Staff	ff. Tr. 8775
		Roger Blond	Section Leader for the Accident Risk Section of the Reactor Risk Branch of the Division of Risk Analysis of the Office of Research - NRC	NRC Staff	ff. Tr. 8775

LIST OF WITNESSES - QUESTION 1

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Sears Testimony	Testimony of John R. Sears of the NRC Staff on Contention 1.1 Under Commission Question 1	John R. Sears	Senior Reactor Safety Engineer in the Emergency Preparedness Licensing Branch, Division of Emergency Preparedness, Office of Inspection and Enforcement - NRC	NRC Staff	ff. Tr. 8692
Sholly 1	UCS/NYPIRG Testimony of Steven C. Sholly on the Consequences of Accidents at Indian Point (Commission Question One and Board Question 1.1)	Steven C. Sholly	Technical Research Associate with the Union of Concerned Scientists	UCS/NYPIRG	ff. Tr. 8398
Soffer Testimony	Direct Testimony of Leonard Soffer Regarding Population Information	Leonard Soffer	Section Leader of the Site Analysis Section, Siting Analysis Branch, Office of Nuclear Reactor Regulation - NRC	NRC Staff	ff. Tr. 8571
Stratton, et al.	Licensees Testimony of William R. Stratton, Walton A. Rodger, and Thomas E. Potter on Question One	William R. Stratton	Independent Consultant	Con Edison/ PASNY	ff. Tr. 8190
		Walton A. Rodger	Chairman of the Board of OTHA, Inc.	Con Edison/ PASNY	ff. Tr. 8190
		Thomas E. Potter	Consultant at Pickard, Lowe and Garrick, Inc. in Public Health Consequence Analysis of Radioactive Releases	Con Edison/ PASNY	ff. Tr. 8190

LIST OF WITNESSES - QUESTION 1

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Weatherwax Testimony	UCS/NYPIRG Testimony of Robert K. Weatherwax on the Indian Point Probabilistic Safety Study	Robert K. Weatherwax	President of Sierra Energy and Risk Assessment, Inc.	UCS/NYPIRG	ff. Tr. 7918
Woods/ Klecker Testimony	Dr. Hugh W. Woods and Raymond W. Klecker on Board Question 1.4	Dr. Hugh W. Woods	NRC Task Manager for the Pressurized Thermal Shock Unresolved Safety Issue	NRC Staff	ff. Tr. 8700
		Raymond W. Klecker	Section Leader, Materials Engineering Branch, Division of Engineering - NRC	NRC Staff	ff. Tr. 8700
No title	December 30, 1982 letter from Dr. Donald A. Wesley to Mr. H. F. Perla	Dennis C. Bley	Consultant With Pickard, Lowe and Garrick in Reliability, Risk, and Decision Analysis for Electric Generating Plants	Con Edison/ PASNY	ff. Tr. 12,985
		Harold F. Perla	Associate Consultant With Pickard, Lowe and Garrick	Con Edison/ PASNY	ff. Tr. 12,985
		Donald A. Wesley	Vice President of Structural Mechanical Associates	Con Edison/ PASNY	ff. Tr. 12,985

LIST OF WITNESSES - QUESTION 2

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Bley 2.2(a)	Licensees' Testimony of Dennis C. Bley on Contention 2.2(a)	Dennis C. Bley	Consultant in Reliability, Risk, and Decision Analysis for Electrical Generating Plants at Pickard, Lowe and Garrick, Inc.	Con Edison/ PASNY	ff. Tr. 6421
Bley Supp. 2.2(a)	Licensees' Supplemental Testimony of Dennis C. Bley on Contention 2.2(a)	Dennis C. Bley	Consultant in Reliability, Risk, and Decision Analysis for Electrical Generating Plants at Pickard, Lowe and Garrick, Inc.	Con Edison/ PASNY	ff. Tr. 13,067
Bley/ Richardson 2	Licensees' Testimony of Dennis C. Bley and Dennis C. Richardson on Commission Question 2 and Board Question 2.2.1	Dennis C. Bley	Consultant in Reliability, Risk, and Decision Analysis for Electrical Generating Plants at Pickard, Lowe and Garrick, Inc.	Con Edison/ PASNY	ff. Tr. 6442
		Dennis C. Richardson	Risk Assessment Technology Manager in the Nuclear Safety Department of the Nuclear Technology Division of Westinghouse Electric Corporation	Con Edison/ PASNY	ff. Tr.6442

LIST OF WITNESSES - QUESTION 2

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Bley/ Richardson 2.1	Licensees' Testimony of Dennis C. Bley and Dennis C. Richardson on Contentions 2.1(a) and 2.1(d)	Dennis C. Bley	Consultant in Reliability, Risk, and Decision Analysis for Electrical Generating Plants at Pickard, Lowe and Garrick, Inc.	Con Edison/ PASNY	ff. Tr. 6280
		Dennis C. Richardson	Risk Assessment Technology Manager in the Nuclear Safety Department of the Nuclear Technology Division of Westinghouse Electric Corporation	Con Edison/ PASNY	ff. Tr. 6280
Bley/ Richardson 2.2.1	Licensees' Testimony of Dennis C. Bley and Dennis C. Richardson on Commission Question 2 and Board Question 2.2.1	Dennis C. Bley	Consultant in Reliability, Risk, and Decision Analysis for Electrical Generating Plants at Pickard, Lowe and Garrick, Inc.	Con Edison/ PASNY	ff. Tr. 6442
		Dennis C. Richardson	Risk Assessment Technology Manager in the Nuclear Safety Department of the Nuclear Technology Division of Westinghouse Electric Corporation	Con Edison PASNY	ff. Tr. 6442

LIST OF WITNESSES - QUESTION 2

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Brons/Josiger Testimony	Power Authority's Testimony of John C. Brons and William Josiger on Board Question 2.2.1	John C. Brons	Resident Manager of the Indian Point Unit 3 Nuclear Power Plant	PASNY	ff. Tr. 6055
		William Josiger	Superintendent of Power of the Indian Point Unit 3 Nuclear Power Plant	PASNY	ff. Tr. 6055
Chapple, et al.	Power Authority's Testimony of John C. Brons, Kenneth R. Chapple, and William Spataro on Contention 2.2(a)	John C. Brons	Resident Manager of the Indian Point Unit 3 Nuclear Power Plant	PASNY	ff. Tr. 6537
		Kenneth R. Chapple	Acting Director, Nuclear Operations and Maintenance of Indian Point Unit 3	PASNY	ff. Tr. 6537
		William Spataro	Senior Metallurgist, Mechanical Design and Analysis Department of the Power Authority	PASNY	ff. Tr. 6537

LIST OF WITNESSES - QUESTION 2

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Fleisher 2.2(a)	West Branch Conservation Association Testimony of Walter L. Flesiher, Contention 2.2(a)	Walter L. Fleisher	Vice President of West Branch Conservation Association	WBCA	ff. Tr. 6492
Fleisher 2.2.1	Testimony of Walter L. Fleisher, Contention 2.2.1	Walter L. Fleisher	Vice President of West Branch Conservation Association	WBCA	ff. Tr. 6493
Holahan/ Rowsome Testimony	Testimony of Gary M. Holahan and Frank H. Rowsome, III on Board Question 2.2.1	Gary M. Holahan	Section Leader in the Operating Reactors Assessment Branch, Division of Licensing - NRC	NRC Staff	ff. Tr. 6919
		Frank H. Rowsome, III	Deputy Director of the Division of Risk Analysis in the Office of Research - NRC	NRC Staff	ff. Tr. 6919
LeFave, et al	NRC Staff Testimony of William T. LeFave, Frank Rowsome and Bernard Turovlin on Contention 2.2.a	William T. LeFave	Senior Auxiliary Systems Engineer in the Auxiliary Systems Branch of the Division of Systems Integration, Office of Nuclear Reactor Regulation - NRC	NRC Staff	ff. Tr. 6890

LIST OF WITNESSES - QUESTION 2

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
LeFave, et al (Continued)	NRC Staff Testimony of William T. LeFave, Frank Rowsome and Bernard Turovlin on Contention 2.2.a	Frank Rowsome	Deputy Director of the Division of Risk Analysis in the Office of Nuclear Regulatory Research - NRC	NRC Staff	ff. Tr. 6890
		Bernard Turovlin	Corrosion Engineer in the Chemical Engineering Branch of the Office of Nuclear Reactor Regulation - NRC	NRC Staff	ff. Tr. 6890
Lee Testimony	Con Edison's Testimony of Min L. Lee on Low Leakage Loading Pattern	Min L. Lee	Chief Nuclear Engineer at Con Edison	Con Edison	ff. Tr. 13,086
Meyer 2.1	Direct Testimony of James F. Meyer Concerning Contention 2.1 (a and d)	James F. Meyer	Senior Task Manager Responsible for the Reactor System and Containment System Portions of Severe (Coremelt) Accident Analysis for Risk Assessment - NRC	NRC Staff	ff. Tr. 6692

LIST OF WITNESSES - QUESTION 2

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Rothstein Testimony	Con Edison's Testimony of Samuel Rothstein Concerning Board Question 2.2.1	Samuel Rothstein	Head of the Chemical and Metallurgical Engineering Subsection in the Mechanical Studies Section at Con Edison	Con Edison	ff. Tr. 6104
Rothstein/ Tuthill Testimony	Con Edison's Testimony of Samuel Rothstein and Arthur Tuthill Concerning Contention 2.2(a)	Samuel Rothstein	Head of the Chemical and Metallurgical Engineering Subsection in the Mechanical Studies Section at Con Edison	Con Edison	ff. Tr. 6515
		Arthur Tuthill	Metallurgical Consultant	Con Edison	ff. Tr. 6515
Rowsome/ Hannon 2	Direct Testimony of Frank Rowsome and John N. Hannon Concerning Commission Question 2	Frank Rowsome	Deputy Director of the Division of Risk Analysis in the Office of Nuclear Regulatory Research - NRC	NRC Staff	ff. Tr. 6563

LIST OF WITNESSES - QUESTION 2

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Rowsome/ Hannon 2 (Continued)	Direct Testimony of Frank Rowsome and John N. Hannon Concerning Commission Question 2	John N. Hannon	Project Manager in the Operating Reactors Branch No. 1 of the Division of Licensing, Office of Nuclear Reactor Regulation - NRC	NRC Staff	ff. Tr. 6563
Sholly/ Thompson Testimony	UCS/NYPIRG Testimony of Gordon R. Thompson and Steven C. Sholly on Commission Question Two, Contentions 2.1(a) and 2.1(d)	Gordon R. Thompson	Senior Staff Scientist with the Union of Concerned Scientists	UCS/NYPIRG	ff. Tr. 6147
		Steven C. Sholly	Technical Research Associate with the Union of Concerned Scientists	UCS/NYPIRG	ff. Tr. 6147

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Altschuler Testimony	Testimony of Richard Altschuler	Richard J. Altschuler	Sociologist, Statistician, Teacher, Writer and Editor	UCS/NYPIRG, RCSE	ff. Tr. 10,880
Ancona Testimony	Testimony of Helga Ancona on Behalf of RCSE/WBCA Relating to Contentions 3.1, 3.2, 3.3, 3.4, 3.7, 3.9, 4.1, 4.2, 4.3 May 1982	Helga Ancona	Journalist and Parent, Stony Point	RCSE, WBCA, Parents, UCS/NYPIRG	ff. Tr. 10,781
Anderson Testimony	Considerations for Planning the Evacuation of Transportation-Handicapped Persons from New York City in the Event of a Life Threatening Nuclear Hazard	Samuel W. Anderson	President of the New York State Coalition of People With Disabilities	UCS/NYPIRG	ff. Tr. 9,482
Anderson Supplemental Testimony	Considerations for Planning the Evacuation of Transportation-Handicapped Persons from New York City - Supplemental Testimony	Samuel W. Anderson	President of the New York State Coalition of People With Disabilities	UCS/NYPIRG	ff. Tr. 9,482
Anderson Supplemental Testimony	Considerations for Planning the Evacuation of Transportation Handicapped Persons from New York City --- Corrections and Additions to Resume	Samuel W. Anderson	President of the New York State Coalition of People with Disabilities	UCS/NYPIRG	ff. Tr. 9,482

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Awalt Testimony	Charles Awalt June 1982 Supplemental Charles Awalt February 1983	Charles Awalt	Community Residence Specialist for the Westchester Developmental Disabilities Services Office of New York State Office of Mental Retardation and Developmental Disabilities, Tarrytown, New York	Parents, WESPAC	ff. Tr. 10,524
Baff Testimony	Shareane Baff	Shareane Baff	Director, Yorktown Gymnastic Center; Parent	Parents	ff. Tr. 9999
Bates Testimony	Testimony of Westchester County Commissioner Charles W. Bates, Department of Social Services	Charles W. Bates	Commissioner of Social Services for Westchester County	Westchester County	ff. Tr. 5658
Bates Supplemental Testimony	Supplemental Testimony of Commissioner Charles W. Bates, Department of Social Services	Charles W. Bates	Commissioner of Social Services for Westchester County	Westchester County	ff. Tr. 5658
Bergman Testimony	Betsy Bergman	Betsy Bergman	Administrative Assistant, Keon Center of Community Aid for Retarded Citizens; Parent	WESPAC	ff. Tr. 10,524

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Bohlander Testimony	Testimony of Westchester County Commissioner Frank C. Bohlander, Department of Public Works	Frank C. Bohlander	Commissioner of Public Works of the County of Westchester	Westchester County	ff. Tr. 5726
Bower Testimony	Testimony of Richard H. Bower, Volunteer Ambulance Disaster Coordinator, Rockland County	Richard Bower	Volunteer Ambulance Disaster Coordinator, Rockland County	WBCA	ff. Tr. 11,103
Bowles Testimony	Raymond Bowles	Raymond Bowles	Teacher, Speech Pathologist, Pocantico Hills School; Parent	Parents	ff. Tr. 10,138
Brooker Testimony	Testimony of Stanley G. Brooker on Behalf of WBCA Relating to Contentions 3.1, 3.2, 3.3, 3.9, 4.3, 4.4, 4.5, May 1982	Stanley G. Brooker	Member, Stony Point Ambulance Corps	WBCA	ff. Tr. 10,225
Burger Testimony	Testimony of Gladys Burger, President, Disabled in Action	Gladys Burger	President, Disabled in Action; Resident, Spring Valley	WBCA, RCSE	ff. Tr. 10,401

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Burnham Testimony	Helen Burnham	Helen Burnham	Director, Croton Free Library	Parents	ff. Tr. 9997
L. Cohen Testimony	Testimony of Lester A. Cohen in Rebuttal to Testimony of Daniel Gutman	Lester A. Cohen	Senior Meterologist, Con Edison	Con Edison	ff. Tr. 15,197
O. Cohen Testimony	Testimony of Oscar Cohen, 20 Burd Street, Nyack, New York Relating to Contention 4.4	Oscar Cohen	Principal, Lexington School for the Deaf, Jackson Heights, New York	Parents, WBCA	ff. Tr. 10,020
Collins Testimony	Testimony of Chief George Schnakenberg, Chief of Police, Town of Clarkstown, N.Y.	William Collins	Chief of Police of Town of Clarkstown (Schnakenberg)	WBCA	ff. Tr. 10,804
Con Edison Onsite Testimony	Con Edison's Testimony in Regard to On-Site Emergency Planning	Lester A. Cohen	Senior Meteorologist, Con Edison	Con Edison	ff. Tr. 11,713
		Charles M. Jackson	Vice President, Nuclear, Con Edison	Con Edison	ff. Tr. 11,713
Con Edison Supplemental Onsite Testimony	Con Edison's Supplemental Testimony of Lester A. Cohen, Charles W. Jackson, George Liebler and William A. Monti Concerning On-Site Emergency Planning	George Liebler	Emergency Planning Director, IP 2, Con Edison	Con Edison	ff. Tr. 11,713
		William A. Monti	Nuclear Consultant - Power Generation, Con Edison	Con Edison	ff. Tr. 11,713

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
L. Concklin Testimony	Testimony of Honrable Lucien H. Concklin, Supervisor, Town of Stony Point	Lucien H. Concklin	Supervisor, Town of Stony Point	WBCA, RCSE	ff. Tr. 10,246
C. Conklin Testimony	Testimony of Cleland S. Conklin on Behalf of RCSE/ WBCA	Cleland S. Conklin	Superintendent of Buildings, Grounds and Transportation, South Orangetown School District	Parents, UCS/NYPIRG	ff. Tr. 10,126
Connelly Testimony	Statement to the Licensing and Safety Board of the NRC	Edward J. Connelly	Emergency Medical Technician, and Member, Ossining Volunteer Ambulance Corps, Ossining	Parents, UCS/NYPIRG	ff. Tr. 10,048
Cormican Testimony	Testimony for Nuclear Regulatory Commission Hearings	Alma Cormican	President, White Plains Teachers Association	UCS/NYPIRG	ff. Tr. 10,076
Corwin Testimony	Seth Corwin	Seth Corwin	President, Chappaqua Bus Co., Inc.	Parents, UCS/NYPIRG	ff. Tr. 10,838
Courtney Testimony	Testimony of Jane Courtney on Behalf of RCSE/WBCA/PARENTS Relating to Contentions 3.1, 3.2, 3.3, 3.7, 3.9, 4.2, 4.3 June 1982	Jane Courtney	Parent	RCSE, WBCA, Parents	ff. Tr. 9701
Craig Testimony	Agata Craig	Agata Craig	Parent	UCS/NYPIRG	ff. Tr. 9681

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Culpepper Testimony	Observations and Conclusions from the March 9, 1983 Drill of the Indian Point Nuclear Power Plants at the Croton-on-Hudson Police Department	Lee Culpepper	Observer	UCS/NYPIRG	ff. Tr. 14,624
Curran Testimony	Testimony of Westchester County Commissioner Dr. Anita S. Curran - Department of Health	Dr. Anita Curran	Commissioner of Health for the County of Westchester	Westchester County	ff. Tr. 5043
Curran Supplemental Testimony	Supplemental Testimony for Indian Point Hearings - Anita S. Curran, M.D., M.P.H., Commissioner of Health, Westchester County, Department of Health				

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Davidoff/Czech Testimony	Testimony of New York State's Witnesses on Commission Questions Three and Four, Donald B. Davidoff, Director, Radiological Emergency Preparedness Group, Lawrence B. Czech, Chief, Nuclear Protection Planning Radiological Emergency Preparedness Group	Donald B. Davidoff	Director, Radiological Emergency Preparedness Group, State of New York	New York State	ff. Tr. 11,313
		Lawerence B. Czech	Chief, Nuclear Protection Planning, Radiological Emergency Preparedness Group	New York State	ff. Tr. 11,313
	Supplemental Testimony of New York State's Witnesses on Commission Questions Three and Four, Donald B. Davidoff, Director, Radiological Emergency Preparedness Group, Lawrence B. Czech, Chief, Nuclear Protection Planning Radiological Emergency Preparedness Group	Lawrence B. Czech	Chief, Nuclear Protection Planning, Radiological Emergency Preparedness Group	New York State	ff. Tr. 11,313
de Ward Testimony	Testimony of Sylvia G. de Ward	Sylvia G. de Ward	Artist, Art Teacher and Parent, Croton-on-Hudson	Parents, WESPAC	ff. Tr. 10,754

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
DelBello Testimony	Testimony of Westchester County Executive Alfred B. DelBello	Alfred DelBello	Former Westchester County Executive and Current Lieutenant Governor of the State of New York	Westchester County	ff. Tr. 5865
DelBello Supplemental Testimony	Supplemental Testimony of Alfred B. DelBello, Westchester County Executive, Lieutenant Governor-Electr, State of New York	Alfred Delbello	Former Westchester County Executive and Current Lieutenant Governor of the State of New York	Westchester County	ff. Tr. 5865
	Amended Testimony of Alfred B. DelBello, Lieutenant Governor, State of New York, January 14, 1983	Alfred Delbello	Former Westchester County Executive and Current Lieutenant Governor of the State of New York	Westchester County	ff. Tr. 5865
del Pilar Testimony	Testimony of: Luis del Pilar, Rockland Resident on Behalf of: RCSE Relating to Contentions: 3.1, 4.7	Luis del Pilar	Affirmative Action Coordinator for the County of Rockland	RCSE	ff. Tr. 10,255
Doughty	Lynn Doughty	Lynn Doughty	President, Croton Harmon Board of Education	Parents	ff. Tr. 9898

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Duffee Testimony	Testimony on the Response of Peekskill Hospital to the Emergency Evacuation Drill, March 9, 1983 by Richard Duffee	Richard Duffee	Observer	UCS/NYPIRG	ff. Tr. 14,607
DuPont Testimony	Power Authority's Testimony of Robert L. DuPont, M.D. on Contention 3.2	Robert L. DuPont	Clinical Professor of Psychiatry at Georgetown Univeristy Medical School; Clinical Associate Professor, Visiting Professor at Harvard Medical School, President of the Phobia Society of America and in private practice of psychiatry	PASNY	ff. Tr. 8852
Dynes Testimony	Licensees' Testimony of Dr. Russell Dynes	Russell R. Dynes	Executive Officer of the American Sociological Association; Member of the American Academy of Political and Social Sciences, the North Central Sociological Association, and the International Sociological Association	Con Edison/ PASNY	ff. Tr. 11,936
Ellefsen Testimony	Earle R. Ellefsen	Earle R. Ellefsen	Captain, Ossining Volunteer Ambulance Corps, Inc.; Parent	Parents	ff. Tr. 10,048

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Elliot Testimony	Statement of Supervisor Nancy R. Elliot, Town of Yorktown	Nancy Elliot	Supervisor, Town of Yorktown	WESPAC, RCSE	ff. Tr. 10,874
Erikson Testimony	Testimony of Kai T. Erikson	Kai T. Erikson	Professor of Sociology and American Studies, Yale University	UCS/NYPIRG	ff. Tr. 9563
Erikson Supplemental Testimony	Supplemental Testimony of Dr. Kai T. Erikson, Professor of Sociology, Yale University, on the Evacuation Plan for Indian Point	Kai T. Erikson		UCS/NYPIRG	ff. Tr. 9563
Everhart Testimony	Testimony of Dr. A. Glen Everhart, Superintendent of Schools North Rock Central School District	A. Glen Everhart	Superintendent of Schools for the North Rockland Central School District	Rockland County	ff. Tr. 4390
Feit Testimony	Observations and Conclusions from the March 9, 1983 Drill of the Indian Point Nuclear Power Plants at the Croton-on-Hudson Police Department	Katerine Feit	Observer	UCS/NYPIRG	ff. Tr. 14,624

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
FEMA Testimony	Direct Testimony of Philip McIntire, Ihor Husar and Joseph H. Keller Concerning Commission Questions 3 and 4	Philip McIntire	Acting Chief of the Natural and Technological Hazards Division of the Region II Office of the Federal Emergency Management Agency (FEMA)	NRC Staff	ff. Tr. 1307
		Ihor Husar	Program Manager for Radiological Emergency Preparedness, Natural and Technological Hazards Division, Region II, FEMA	NRC Staff	ff. Tr. 1307
		Joseph H. Keller	Scientist for Allied Chemical Corporation and Exxon Nuclear Idaho Company	NRC Staff	ff. Tr. 1307

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
FEMA Supplemental Testimony	FEMA's Supplemental Testimony of February 21, 1983 Concerning Commission Questions 3 and 4	Roger Kowieski	Chairman of the Regional Assistance Committee, Natural and Technological Hazards Division, Region II	NRC Staff	ff. Tr. 14,741
		Philip McIntire	Acting Chief of the Natural and Technological Hazards Division of the Region II Office of the Federal Emergency Management Agency	NRC Staff	ff. Tr. 14,720
		Joseph H. Keller	Scientist for Allied Chemical Corporation and Exxon Nuclear Idaho Company	NRC Staff	ff. Tr. 14,720
Fischer Testimony	Orange and Rockland Utilities, Inc., Direct Testimony of Frank E. Fischer	Frank E. Fischer	Vice President of Engineering and Production, Orange and Rockland Utilities, Inc.	Rockland County	ff. Tr. 11,048
Flicker Testimony	Testimony of Dr. Bernard Flicker Supplemental Testimony Rockland County	Bernard Flicker	Professor, Hunter College and Herbert H. Lehman College and Member of the Rockland County Citizens Advisory Committee	Rockland County	ff. Tr. 11,243

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Ford Testimony	Karen K. Ford	Karen K. Ford	Parent	Parents, WESPAC	ff. Tr. 9691
Galdone Testimony	Jannelise Galdone	Jannelise Galdone	Member of West Branch Conservation Association; housewife and photographer	UCS/NYPIRG	ff. Tr. 9638
Gdanski Testimony	Testimony of Sam Gdanski	Sam Gdanski	Member of the Rockland County Legislature and Chairman of the Rockland County Legislature Sub-Committee to review the Indian Point Emergency Response Plan	Rockland County	ff. Tr. 3369
Gendron Testimony	Intervenor-Observer Observation Point	Richard E. Gendron	Observer	UCS/NYPIRG	ff. Tr. 14,559
Gersony Testimony	Toby Gersony	Toby Gersony	Educational Director, Ramaguois Country Day School, Pomona	Parents, RCSE	ff. Tr. 10,711
Gochman Testimony	Barbara Gochman	Barbara Gochman	Teacher, Carrie E. Tompkins Elementary School, Croton-on-Hudson, NY	Parents	ff. Tr. 10,436
Gohring Testimony	Testimony of Mary Lou Gohring, B.S., M.A., Member of the Faculty, North Rockland H.S. on Behalf of RCSE/Parents Relating to Contentions 3.1, 3.2, 3.7, May 1982	Mary Lou Gohring	Teacher, North Rockland High School	RCSE, Parents	ff. Tr. 10,445

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Goldfarb Testimony	Statement of Ronald A. Goldfarb, Chief of Police Village of Ossining, New York Representing Westchester County Chiefs of Police Association	Ronald A. Goldfarb	Chief of the Village of Ossining Police Department	Westchester County	ff. Tr. 5774
Goldfarb Supplemental Testimony	Testimony of Ronald Goldfarb, Chief of Police, Village of Ossining	Ronald A. Goldfarb	Chief of Police, Village of Ossining	WFSPAC	ff. Tr. 9821
Gromack Testimony	Testimony of: Alex Gromack on Behalf of: RCSE/Parents/ WESPAC Relating to Contentions: 3.1, 3.2, 3.6, 3.7, 4.4, 4.7	Alex Gromack	Senior Recreation Leader, Town of Clarkstown Parks and Recreation Department, and member of Transportation Advisory Council of Rockland County	RCSE, Parents	ff. Tr. 10,177
Guido Testimony	Testimony of Westchester County Commissioner/Sheriff Daniel P. Guido, Department of Public Safety	Daniel P. Guido	Commissioner/Sheriff of Public Safety of Westchester County	Westchester County	ff. Tr. 4913
Guido Supplemental Testimony	Supplemental Testimony of Commissioner/ Sheriff Daniej Guido, Department of Public Safety	Daniel P. Guido	Commissioner/Sheriff of Public Safety of Westchester County	Westchester	ff. Tr. 4913

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Gunn Testimony	Shirley S. Gunn	Shirley S. Gunn	Director-Teacher, Circle School; Parent, Croton-on-Hudson	Parents, WESPAC	ff. Tr. 10,702
Gurin, et al.	Testimony Submitted on Behalf of "New York City Council" Intervenors By David Gurin, Deputy Commissioner Joel Friedman, P.E., Robert McCarthy of the New York City Department of Transportation	David Gurin	Deputy Commissioner for Planning, New York City Department of Transportation	New York City Council Members	ff. Tr. 9264
		Joel Friedman	Chief, Plans and Surveys, New York City Department of Transportation	New York City Council Members	ff. Tr. 9264
		Robert McCarthy	Director of Transportation Research, New York City Department of Transportation	New York City Council Members	ff. Tr. 9264
Gutman Testimony	Testimony Submitted on Behalf of UCS/ NYPIRG and Members of the New York City Council by Daniel Gutman	Daniel Gutman	Consultant in the Fields of Air Pollution Dispersion and Traffic Analysis	UCS/NYPIRG	ff. Tr. 15,164
Hare Testimony	Testimony for Submission to the Nuclear Regulatory Commission in its Hearings, Summer 1982, Respecting the Indian Point Nuclear Plant	Robert W. Hare, Pastor	Pastor, Scarborough Presbyterian Church; Parent	Parents, UCS/NYPIRG	ff. Tr. 10,388

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Hastings Testimony	Supplementary Testimony, Rockland County, Donald Hastings	Donald Hastings	Director of Emergency Service for the County of Rockland	Rockland County	ff. Tr. 11,022
Helbraun Testimony	Testimony of Phyllis Helbraun Executive Director of the Rockland Council for Young children on Behalf of RCSE/ Parents Relating to Contentions 3.1, 3.2, 3.7, 4.5, 4.7	Phyllis Helbraun	Executive Director, Rockland Council for Young Children	RCSE	ff. Tr. 10,711
Helbraun Supplemental Testimony	Supplemental Testimony of Phyllis Helbraun, Executive Director of the Rockland Council for Young Children on Behalf of RCSE	Phyllis Helbraun	Executive Director, Rockland Council for Young Children	RCSE	ff. Tr. 10,727
Hickernell Testimony	Barbara K. Hickernell	Barbara K. Hickernell	Observer	UCS/NYPIRG	ff. Tr. 14,532
Holland Testimony	Testimony of Chief Michael Holland, Village of Haverstraw, New York	Michael Holland	Chief of Police of the Village of Haverstraw and Chief of the Mid-Hudson Association of Chiefs of Police	WBCA	ff. Tr. 1661
Holzer Testimony	Madeline and Marc Holzer	Madeline and Marc Holzer	Parents, Croton-on-Hudson	Parents, WESPAC	ff. Tr. 10,731

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Iurato Testimony	John Iurato	John Iurato	Assistant Principal, Croton Harmon High School	Parents, WESPAC	ff. Tr. 9898
Jessup Testimony	Testimony of: Lois Jessup, Rockland Resident on Behalf of: R.C.S.E./ Parents Relating to Contentions: 3.1, 3.2, 3.3, 3.7, 4.4	Lois Jessup	Retired Teacher, E. Ramapo School District and President of the Rockland County Conservation Association	RCSE, Parents	ff. Tr. 10,791
A. Johnson Testimony	Testimony of Albert A. Johnson, Principal, Farley Middle School, Board of Directors, Stony Point Ambulance Corps, Contentions 3.2, 3.3, 3.7, 3.9, 4.3 and 4.4	Albert A. Johnson	Resident, Stony Point	UCS/NYPIRG, Parents	ff. Tr. 10,218
F. Johnson Testimony	Testimony of the Rev. Frederick F. Johnson, Rector on Behalf of WBCA Relating to Contentions 3.1, 3.2, 3.3, 3.9, 4.2, 4.3 May 1982	The Rev. Frederick F. Johnson	Rector of St. Paul's Episcopal Church, Spring Valley, and Member of Board of Governors of Rockland County Health Complex	WBCA	ff. Tr. 10,388

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Jurkowski Testimony	Testimony of Westchester County Deputy Commissioner of Transportation Raymond S. Jurkowski on the Evacuation Plan for Indian Point, given before the Nuclear Regulatory Commission in June/July 1982	Raymond S. Jurkowski	Deputy Commissioner of Transportation for Westchester County	Westchester County	ff. Tr. 5212
Jurkowski Supplemental Testimony	Supplemental Testimony of Westchester County Deputy Commissioner of Transportation, Raymond S. Jurkowski, on the Evacuation Plan for Indian Point	Raymond S. Jurkowski	Deputy Commissioner of Transportation for Westchester County	Westchester County	ff. Tr. 5212
Kahn Testimony	Testimony of Eleanor Kahn, Bittersweet Lane, Mt. Kisco, New York 10549	Eleanor Kahn	Special Education Teacher, Robert E. Bell Middle School; Parent, Chappaqua	Parents, UCS/NYPIRG	ff. Tr. 10,733

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Kaminski Testimony	Testimony of Westchester County Director of Planning, Michael Kaminski, Department of Hospitals	Michael S. Kaminski	Director of Planning for the Westchester County Medical Center	Westchester County	ff. Tr. 5567
Kaminski Supplemental Testimony	Additional Testimony of Michael S. Kaminski	Michael S. Kaminski	Director of Planning for the Westchester County Medical Center	Westchester County	ff. Tr. 11,617
Kesselman Testimony	Testimony of: Judy Kesselman, Coordinator of Suburban Schools, Tiorati Workshop on Behalf of: RCSE/Parents Relating to Contentions: 3.1, 3.7, 4.4, 4.7	Judy Kesselman	Coordinator, Suburban Schools, Tiorati Workshop, Harriman State Park	WBCA, Parents	ff. Tr. 10,001
King Testimony	Testimony of Joan Harding King	Joan Harding King	Acting Officer of West Branch Conservation Association; Housewife; Registered Nurse	UCS/NYPIRG	ff. Tr. 9645
Kinoy Testimony	Testimony Submitted on Behalf of "New York City Council" Intervenors, By Susan K. Kinoy	Susan K. Kinoy	Associate Executive Director for Program, Community Council of Greater New York	New York City Council Members	ff. Tr. 9391

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Kralik Testimony	Testimony of James F. Kralik	James F. Kralik	Chief Deputy Sheriff of the Patrol Division, Rockland County Sheriff's Department, New City	Rockland County	ff. Tr. 3577, 4503
Kralik Supplemental Testimony	Supplemental Testimony Rockland County	James F. Kralik	Chief Deputy Sheriff Rockland County	Rockland County	ff. Tr. 11,132
Lavelle Testimony	Testimony of Myles Lavelle, Councilman, Town of Stony Point, N.Y. on Behalf of RCSE/WBCA Relating to Contentions 3.1, 3.2, 3.6, May 1982	Myles Lavelle	Councilman, Town of Stony Point	RCSE, WBCA	ff. Tr. 10,160
Le Moullec Testimony	Testimony of: Betty Ramey, WRKL Radio on Behalf of: RCSE Relating to Contentions: 3.4, 3.1	Robert Le Moullec	News Director, WRKL Radio in Pomona, New York	RCSE	ff. Tr. 9665
Lebid Testimony	Enid Lebid (Observer)	Enid Lebid	Observer	UCS/NYPIRG	ff. Tr. 14,580
Lecker Testimony	Licensees' Testimony of Sidney Lecker, M.D.	Sidney Lecker	Psychiatrist (Principally Child)	Con Edison/ PASNY Licensees	ff. Tr. 11,966
Lifton Testimony	Testimony of Robert Jay Lifton, M.D.	Robert Jay Lifton	Professor of Psychiatry, Yale School of Medicine	UCS/NYPIRG	ff. Tr. 12,425

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Linden/ Layefsky Testimony	Enid Linden and Flo Layefsky (Observation)	Enid Linden	Observer	UCS/NYPIRG	ff. Tr. 14,636
		Flo Layefsky	Observer	UCS/NYPIRG	ff. Tr. 14,636
[Littlejohn Testimony]	[None - given orally]	Robert Littlejohn	Commanding Officer of the Police Operations Division, New York City Police Department and Coordinator of the Mayor's Emergency Control Board	New York City Council Members	ff. Tr. 9188
Marasco Testimony	Anthony R. Marasco, Director, Westchester County Office of Disaster And Emergency Services (Department of Public Safety) - For the Past Eight Years	Anthony R. Marasco	Director of the Westchester County Office of Disaster and Emergency Services	Westchester County	ff. Tr. 5388
Marasco Supplemental Testimony	Supplemental Testimony of Anthony R. Marasco	Anthony R. Marasco	Director of the Westchester County Office of Disaster and Emergency Services	Westchester County	ff. Tr. 5388
D. McGuire Testimony	Testimony of Donald P. McGuire	Donald P. McGuire	Deputy Director of the Office of Emergency Serivces for Rockland County	Rockland County	ff. Tr. 3929
J. McGuire Testimony	Testimony of James McGuire, Health Department	James F. McGuire	Radiological Health Specialist for the Rockland County Department of Health	Rockland County	ff. Tr. 3650

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Melbin Testimony	Testimony of Murray Melbin, Ph.D., Comment on Emergency Plan for Indian Point Power Plant	Murray Melbin	Professor of Sociology, Boston University	UCS/NYPIRG	ff. Tr. 9944
Mills Testimony	Intervenor Observer Observation Post	Nathaniel Mills	Observer	UCS/NYPIRG	ff. Tr. 14,559
Morris Testimony	Testimony of Robert L. Morris on Behalf of the Union of Concerned Scientists and the New York Public Interest Research Group, Inc. Relating to Board Contentions 3.1, 3.3, 3.6, June 21, 1982	Robert L. Morris	Registered Professional Engineer and Traffic Consultant	UCS/NYPIRG	ff. Tr. 9726
A. Murphy Testimony	Nuclear Regulatory Commission Evacuation If Needed By Indian Point Problem	Agnes Murphy	Program Coordinator, Senior Action Center, Ossining	Parents, WESPAC	ff. Tr. 10,754
J. Murphy Testimony	Testimony of James L. Murphy, MPH, Public Health Specialist, New York Public Interest Research Group, Inc., White Plains, NY	James L. Murphy	Public Health Specialist, New York Public Interest Research Group, Inc., White Plains	UCS/NYPIRG	ff. Tr. 11,060

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Narod-Shiek Testimony	Fern Narod-Shiek	Fern Narod-Sheik	School Bus Driver, Mahopac School District, Westchester County	Parents, UCS/NYPIRG	ff. Tr. 10,838
Northrup Testimony	Testimony of Ruth Northrup	Ruth Northrup	Deputy Commissioner, Department of Social Services, County of Rockland	Rockland County	ff. Tr. 4201
Northrup Supplemental Testimony	Supplemental Testimony, Rockland County	Ruth Northrup	Deputy Commissioner, Department of Social Services, County of Rockland	Rockland County	ff. Tr. 11,279
O'Brien Testimony	Richard Lang	Maureen O'Brien	Employee, Search for Change, Inc.	Parents, WESPAC	ff. Tr. 10,524
O'Lear Testimony	Testimony of George O'Lear for Rockland County	George O'Lear	Associate Planner for the Rockland County Planning Board	Rockland County	ff. Tr. 1808
O'Rourke Testimony	Testimony of Westchester County Executive Andrew P. O'Rourke on Commission Questions Three and Four, March 11, 1983	Andrew P. O'Rourke	Westchester County Executive	Westchester County	ff. Tr. 11,520

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
O'Rourke Supplemental Testimony	Supplemental Testimony of County Executive Andrew P. O'Rourke Before the Nuclear Regulatory Commission - March 24, 1983	Andrew P. O'Rourke	Westchester County Executive	Westchester County	ff. Tr. 11,520
Parsons Testimony	Licensees' Testimony of Parsons Brinckerhoff Quade & Douglas, Inc.	Bruce E. Podwal	Vice President, Parsons, Brinckerhoff Quade & Douglas, Inc.; Senior Professional Associate Deputy Technical Director - Civil/Highways	Con Edison/ PASNY	ff. Tr. 11,774
Parsons Supplemental Testimony	Licensees' Supplemental Testimony of Parsons Brinckerhoff Quade & Douglas, Inc.	Peggy L. Rosenblatt	Senior Urban/Environmental Planner, Parsons Brinckerhoff Quade & Douglas	Con Edison/ PASNY	ff. Tr. 11,774
		Michael S. Della Rocca	Transportation Planning Engineer	Con Edison/ PASNY	ff. Tr. 11,774
Pettersen Testimony	Jack Pettersen	Jack Pettersen	Superintendent, Town of Cortlandt Parks, Recreation and Conservation Department	Parents	ff. Tr. 10,169

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Plage, et al.	Ossining Observation - Plage/ Sanchez/Indusi	Ms. Plage	Observer	UCS/NYPIRG	ff. Tr. 14,532
		Ms. Sanchez	Observer	UCS/NYPIRG	ff. Tr. 14,532
		Ms. Indusi	Observer	UCS/NYPIRG	ff. Tr. 14,532
Poritzky Testimony	Observer: Hal Poritzky, Location: Yorktown Police Department	Hal Poritzky	Observer	UCS/NYPIRG	ff. Tr. 14,500
Powell Testimony	West Branch Conservation Association's Testimony of Neva Powell, March 9 Exercise	Neva Powell	Art Teacher, Rockland Center for the Arts and the Rockland County Center for the Physically Handicapped; Grandmother; Observer	WBCA	ff. Tr. 14,512
Power Authority Onsite Testimony	Power Authority's Testimony of John C. Brons, Linda Lomonaco and Dennis Quinn	John C. Brons	Resident Manager at Indian Point 3	PASNY	ff. Tr. 11,679
		Linda Lomonaco	Assistant to the Radiological Environmental Services Superintendent at Indian Point 3	PASNY	ff. Tr. 11,679
		Dennis Quinn	Radiological Engineer, Radiological and Environmental Services Department at Indian Point 3	PASNY	ff. Tr. 11,679

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Puglisi Testimony	Linda Puglisi (Letter)	Linda Puglisi	Director, Mohegan Colony Nursery School, Crompond	Parents, WESPAC	ff. Tr. 10,711
Richter Testimony	Pat Richter (Statement)	Pat Richter	Resident Supervisor, Putnam Association for Retarded Citizens; Parent	Parents, WESPAC	ff. Tr. 10,524
Reisman Testimony	Supplementary Testimony, Rockland County, Herbert Reisman	Herbert Reisman	Chairman of the Rockland County Legislature	Rockland County	ff. Tr. 11,027
Robock Testimony	Comments on Evacuation Observation by Jerry Robock, Yorktown Police Headquarters, Yorktown Heights, NY, March 9, 1983	Jerry Robock	Observer	UCS/NYPIRG	ff. Tr. 14,500
Rowley Testimony	Statement of Clifford Rowley and Nancy Rowley	Clifford Rowley	President, New York City Civic Association of the Deaf	Parents	ff. Tr. 10,025
		Nancy Rowley	President, Westchester Community Services for the Hearing Impaired	Parents	ff. Tr. 10,025

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Rubeo Testimony	Written Testimony Submitted to the Nuclear Regulatory Commission on June 4, 1982 by Vincent J. Rubeo, Teacher at Anne M. Dorner Middle School, Ossining, New York	Vincent J. Rubeo	Teacher, Anne M. Dorner Middle School, Ossining	Parents	ff. Tr. 10,436
Saunders Testimony	Kathe Saunders (Statement)	Kathe Saunders	Croton Recreation Department After-School Program and Parent	Parents	ff. Tr. 10,928
[Scalpi Testimony]	[None - Given Orally]	Michael Scalpi	Civil Defense Director, Putnam County	Power Authority	ff. Tr. 12,063
Scharf Testimony	Testimony of Leonard R. Scharf	Leonard R. Scharf	Assistant Superintendent for Business and Finance, Rockland County Board of Cooperative Educational Services	Rockland County	ff. Tr. 11,154
[Schmer Testimony]	[None - Given Orally]	Philip Schmer	Assistant Director, Office of Natural Disaster and Civil Defense, Orange County	Power Authority	ff. Tr. 12,063

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Schwartz Testimony	Testimony of Sheldon A. Schwartz, Deputy Director of Emergency Preparedness and Engineering Response, U.S.N.R.C. Concnering Contentions Related to Commission Questions 3 & 4	Sheldon A. Schwartz	NRC Deputy Director, Division of Emergency Preparedness and Engineering Response	NRC Staff	ff. Tr. 12,244
Scurti Testimony	Testimony of Stephen Scurti, Police Chief, Stony Point, N.Y. Concerning Contentions 3.1, 3.2, 3.3, 3.7, 3.9 & 4.3	Stephen Scurti	Chief of Police of Stony Point, New York	WBCA	ff. Tr. 1661
Seasonwein Testimony	Testimony of Roger Seasonwein on Commission Question 3 Proferred by Assemblyman Richard Brodsky	Roger Seasonwein	President, Seasonwein Associates	Brodsky	ff. Tr. 10,586
Sears Testimony	Testimony of John R. Sears of the NRC Staff on Commission Questions 3 and 4 and Emergency Planning Contentions Admitted by Board Order of April 23, 1982 for Indian Point, Unit No. 2 and Unit No. 3	John R. Sears	Senior Reactor Safety Engineer, Emergency Preparedness Licensing Branch, Division of Emergency Preparedness, Office of Inspection and Enforcement, NRC	NRC Staff	ff. Tr. 12,244

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Sekelsky Testimony	Elizabeth Sekelsky (Statement)	Elizabeth Sekelsky	School Nurse, Carrie E. Tompkins Elementary School; Parent	Parents	ff. Tr. 10,437
Seley Testimony	Testimony Submitted on Behalf of "New York City Council", Intervenors, by John E. Seley	John E. Seley	Associate Professor, Department of Urban Studies, Queens College/CUNY	New York City Council Members	ff. Tr. 9333
Schultz Testimony	Testimony of William G. Borghard, P.E. Commissioner of the Department of Environmental Facilities	Richard Schultz	Licensed Professional Engineer, Employed by Westchester County as Deputy Commissioner of Environmental Facilities	Westchester County	ff. Tr. 5824
Siegel Testimony	David Siegel	David Siegel	Superintendent of Schools, Croton-On-Harmon School District	Parents, RCSE, UCS/NYPIRG	ff. Tr. 9898
Simon Testimony	Susan Simon	Susan Simon	Public Health Nurse, Westchester County	WESPAC	ff. Tr. 10,754
Smith Testimony	Testimony of Dr. Don D. Smith	Donald D. Smith	Professor of Mass Communication in the School of Journalism, University of Iowa	UCS/NYPIRG	ff. Tr. 10,269
Solnit Testimony	Testimony of Albert J. Solnit, M.D.	Albert J. Solnit	Sterling Professor of Pediatrics and Psychiatry and Director of the Child Study Center at Yale University	UCS/NYPIRG, Parents	ff. Tr. 10,459

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Solon Testimony	Testimony Submitted on Behalf of "New York City Council" Intervenors by Dr. Leonard Solon	Leonard R. Solon	Director of the Bureau for Radiation Control, New York City Department of Health	New York City Council Members	ff. Tr. 8981
Spiegel Testimony	Testimony of Leonard Spiegel	Leonard Spiegel	Transit Coordinator for Rockland County	Rockland County	ff. Tr. 4600
Spohn Testimony	Testimony Submitted on Behalf of "New York City Council" Intervenors by Roberta R. Spohn, Deputy Commissioner, New York City Department for the Aging	Roberta R. Spohn	Deputy Commissioner of the Department for the Aging of the City of New York	New York City Council	ff. Tr. 9393
Tomkins Testimony	Marie R. Tomkins	Marie R. Tomkins	Resident, Tomkins Cove	WBCA	ff. Tr. 10,118
Urbanik Testimony	Testimony of Dr. Thomas Urbanik, II Concerning the Evacuation Time Estimate Studies for Indian Point, Units 2 and 3	Dr. Thomas Urbanik, II	Assistant Research Engineer associated With the Texas Transportation Institute of the Texas A & M University System, College Station, Texas	NRC Staff	ff. Tr. 1861

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Ward Testimony	Testimony Submitted on Behalf of "New York City Council" Intervenors by Benjamin Ward, Commissioner, New York City Department of Correction	Benjamin Ward	Commissioner of New York City Department of Correction	New York City Council	ff. Tr. 9166
Waterfall Testimony	Beth Waterfall	Beth Waterfall	Press Observer, Croton-Cortlandt News	WESPAC, RCSE	ff. Tr. 10,571
Wayne Testimony	From the Reverend David B. Wayne, Rector of St. Augustine's Episcopal Church, Old Post Road North and Grand Street, Croton-on-Hudson	Rev. David B. Wayne	Rector, St. Augustine's Episcopal Church; Parent, Croton-on-Hudson	Parents	ff. Tr. 10,388
Wein Testimony	Testimony from Rockland County Association for the Visually Impaired, Inc.	Ruth Wein	Director of the Rockland County Association for the Visually Impaired, Inc.	Rockland County	ff. Tr. 4578

LIST OF WITNESSES - QUESTIONS 3 AND 4

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Wexler Testimony	Stephanie Wexler (Observer)	Stephanie Wexler	Observer	UCS/NYPIRG	ff. Tr. 14,587
	Observations of Stephanie Wexler on Wednesday, March 9, 1983 Putnam County Police Station, Putnam County, N.Y.				
Wishnie Testimony	Richard Wishnie	Richard Wishnie	Supervisor, Town of Ossining	WESPAC	ff. Tr. 9820
Ziegler Testimony	Testimony for WESPAC Concerning Emergency Evacuation Plan as it Relates to Children in Day Care	Sally Ziegler	Executive Director, Day Care Council of Westchester, Inc.	WESPAC, Parents, RCSE	ff. Tr. 10,711

LIST OF WITNESSES - QUESTION 5

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
B. Cohen Testimony	Licensees' Testimony of Bernard L. Cohen on Commission Question 5	Bernard L. Cohen	Professor of Physics and of Chemical and Petroleum Engineering at the University of Pittsburgh	Con Edison/ PASNY	ff. Tr. 14,427
Paddelford, et al.	Licensees Testimony of Dennis C. Bley, Donald F. Paddelford, Thomas E. Potter, and Dennis C. Richardson on Commission Question 5	Dennis C. Bley	Consultant at Pickard, Lowe and Garrick, Inc. in Reliability, Risk, and Decision Analysis for Electrical Generating Plants	Con Edison/ PASNY	ff. Tr. 12,663
		Donald F. Paddleford	Advisory Engineer in the Risk Assessment Section of the Nuclear Safety Department of the Nuclear Technology Division of Westinghouse Electric Corporation	Con Edison/ PASNY	ff. Tr. 12,663
		Thomas E. Potter	Consultant at Pickard, Lowe and Garrick, Inc. in Public Health Consequence Analysis of Radioactive Releases	Con Edison/ PASNY	ff. Tr. 12,663
		Dennis C. Richardson	Risk Assessment Technology Manger in the Nuclear Safety Department of the Nuclear Technology Division of Westinghouse Electric Corporation	Con Edison/ PASNY	ff. Tr. 12,663

LIST OF WITNESSES - QUESTION 5

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Potter 5	Licensees' Testimony of Thomas E. Potter on Commission Question Five	Thomas E. Potter	Consultant at Pickard, Lowe and Garrick, Inc. in Public Health Consequence Analysis of Radioactive Releases	Con Edison/ PASNY	ff. Tr. 12,782
Rowsome/ Blond 5	Direct Testimony of Frank Rowsome and Roger Blond Concerning Commission Question 5	Frank Rowsome	Assistant Director for Technology, Division of Systems Technology of the Office of Nuclear Reactor Regulation of the NRC	NRC Staff	ff. Tr. 12,834
		Roger Blond	Section Leader for the Accident Risk Section of the Risk Branch of the Division of Risk Analysis of the Office of Research - NRC	NRC Staff	ff. Tr. 12,834
Sholly 5	UCS/NYPIRG Testimony of Steven C. Sholly on Commission Question Five	Steven C. Sholly	Technical Research Associate With the Union of Concerned Scientists	UCS/NYPIRG	ff. Tr. 12,730

LIST OF WITNESSES - QUESTION 6

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Billups et al.	NRC Panel Regarding the Environmental and Socioeconomic Consequences of a Shutdown of Indian Point Unit 2 and/or Unit 3	Charles W. Billups	Aquatic Scientist of the Environmental Engineering Section in the Environmental and Hydrologic Engineering Branch of the Division of Engineering Within the Office of Nuclear Reactor Regulation of the NRC	NRC Staff	ff. Tr. 14,306
		Germain E. Laroche	Senior Land Use Analyst of the Environmental Engineering Section	NRC Staff	ff. Tr. 14,306
		Edwin D. Pentecost	Senior Land Use Analyst of the Environmental Engineering Section	NRC Staff	ff. Tr. 14,306
		Francis C. Kornegay	Research Staff Member of the Environmental Impact Section in the Energy Division of the Oak Ridge National Laboratory	NRC Staff	ff. Tr. 14,306
		Sam A. Carnes	Member of the Research Staff in the Social Impact Analysis Group of the Energy Division of the Oak Ridge National Laboratory	NRC Staff	ff. Tr. 14,306

LIST OF WITNESSES - QUESTION 6

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Branagan Testimony	NRC Staff Testimony of Edward F. Branagan, Jr. on Contention 6.2	Edward F. Branagan, Jr.	Health Physicist in the Division of Systems Integration, Office of Nuclear Reactor Regulation of the NRC	NRC Staff	ff. Tr. 14,361
Branagan Supplemental Testimony	NRC Staff Supplemental Testimony of Edward F. Branagan on Contention 6.2	Edward F. Branagan, Jr.	Health Physicist in the Division of Systems Integration, Office of Nuclear Reactor Regulation of the NRC	NRC Staff	ff. Tr. 15,143
Buehring et al.	NRC Staff Testimony of William A. Buehring, Sidney E. Feld, Erastace N. Fields, Philip R. Nicholson, and Robert S. Wood Regarding System Reliability and Economic Consequences of a Shutdown of Indian Point Unit 2 and/or Unit 3	William A. Buehring	Energy Systems Engineer in the Decision and System Sciences Section, Argonne National Laboratory	NRC Staff	ff. Tr. 14,130
		Sidney E. Feld	Senior Cost Analyst in the Analysis Group of the Division of Budget and Analysis Within the Office of Resource Management of the NRC	NRC Staff	ff. Tr. 14,130
		Erastace N. Fields	Electrical Engineer in the Site Analysis Branch of the Division of Engineering, Office of Nuclear Reactor Regulation of the NRC	NRC Staff	ff. Tr. 14,130

LIST OF WITNESSES - QUESTION 6

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Buehring et al.	NRC Staff Testimony of William A. Buehring, Sidney E. Feld, Erastace N. Fields, Philip R. Nicholson, and Robert S. Wood Regarding System Reliability and Economic Consequences of a Shutdown of Indian Point Unit 2 and/or Unit 3	Philip R. Nicholson	Antitrust Economist in the Site Analysis Branch of the Division of Engineering Office of Nuclear Reactor Regulation of the NRC	NRC Staff	ff. Tr. 14,130
		Robert S. Wood	Licensee Relations Analyst in the Licensee Relations Section of the Office of State Programs of the NRC	NRC Staff	ff. Tr. 14,130
Commoner/ Schrader Testimony	Testimony Submitted on Behalf of "New York City Council" Intervenors by Dr. Barry Commoner and Mr. Richard Schrader	Barry Commoner	Employed at the Center for the Biology of Natural Systems, Queens College; Member of the editorial Board of the Bulletin of Science Technology and Society; Member of the Advisory Board of the Scientific Advisor, New York State Legislative Commission on Science	New York City Council Members	ff. Tr. 13,951

LIST OF WITNESSES - QUESTION 6

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Commoner/ Schrader Testimony (Continued)	Testimony Submitted on Behalf of "New York City Council" Intervenors by Dr. Barry Commoner and Mr. Richard Schrader	Richard Schrader	Research Associate at the Center for the Biology of Natural Systems, Queens College	New York City Council Members	ff. Tr. 13,951
Conant Testimony	Power Authority's Testimony of Melvin A. Conant on Commission Question 6.3	Melvin A. Conant	Advisor on Political Aspects of Access to Strategic Raw Materials Generally and Oil and Gas Supply in Particular and Chairman of World Bank Energy Policy Group (Korea)	PASNY	ff. Tr. 13,667
Dean et al.	Power Authority's Testimony of Charles R. Dean, Harold M. Hochman and Paul H. Rubin on Commission Question 6.3	Charles R. Dean	Economic Consultant Who Has Worked on Energy Matters	PASNY	ff. Tr. 13,645
		Harold M. Hochman	Professor of Economics and Director of the Center for the Study of Business and Government at Baruch College of the City University of New York	PASNY	ff. Tr. 13,645
		Paul H. Rubin	Professor of Economics and a Research Associate at the Center for the Study of Business and Government at Baruch College of the City University of New York	PASNY	ff. Tr. 13,645

LIST OF WITNESSES - QUESTION 6

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Dunbar Testimony	Licensees' Testimony of Dr. Frederick C. Dunbar on Commission Question 6	Frederick C. Dunbar	Senior Consultant With National Economic Research Associates, Inc. Specializing in the Economics of Energy and Transportation	Con Edison/ PASNY	ff. Tr. 13,604
Fitzpatrick Testimony	Licensees' Testimony of George L. Fitzpatrick on Commission Question 6	George L. Fitzpatrick	President of Applied Energy Group, Inc.	Con Edison/ PASNY	ff. Tr. 14,327
Fleisher 6.1 Testimony	Testimony of Walter L. Fleisher, Contention 6.1	Walter L. Fleisher	Vice President of the West Branch Conservation Association	WBCA	ff. Tr. 14,079
Freudenthal Testimony	Con Edison's Testimony of Dr. Peter C. Freudenthal on Commission Question 6	Peter C. Freudenthal	Director of Air and Noise Programs for Con Edison	Con Edison	ff. Tr. 13,673
Kelly Testimony	Testimony of Larry Kelly - not prefiled	Larry Kelly	Performance Reliability Supervisor at Indian Point 3	PASNY	ff. Tr. 13,591
Meehan Testimony	Licensees' Testimony of Eugene T. Meehan on Commission Question 6	Eugene T. Meehan	Vice President of Energy Management Associates, Inc.	Con Edison/ PASNY	ff. Tr. 13,167

LIST OF WITNESSES - QUESTION 6

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Parmelee Testimony	Testimony of New York State Witness James M. Parmelee	James M. Parmelee	Electric Energy Planner for the New York State Energy Office	New York State	ff. Tr. 13,727
Rosen Testimony	Testimony of Greater New York Council on Energy Witness Richard A. Rosen on Commission Question 6.3	Richard A. Rosen	Senior Research Scientist at Energy Systems Research Group and Vice President of the Firm	Greater New York Council on Energy	ff. Tr. 13,788
Schlissel/ Mavretich Testimony	Testimony Submitted on Behalf of "New York City Council" Intervenors By Mr. David Schlissel and Mr. John Mavretich	David Schlissel	Private Energy Consultant to State Agencies and tc Community and Consumer Organizations	New York City Council Members	ff. Tr. 14,266
		John Mavretich	Research Assistant to Honorable Maurice D. Hinchey, Chairman of the Committee on Environmental Conservation, New York State Assembly	New York City Council Members	ff. Tr. 14,266
Stewart Testimony	Licensees' Testimony of Allan M. Stewart on Commission Question 6	Allan M. Stewart	Director of the Gas Supply Department at Con Edison	Con Edison/ PASNY	ff. Tr. 13,715
Streiter Testimony	Licensees' Testimony of Sally Hunt Streiter on Commission Question 6	Sally Hunt Streiter	Vice President of National Economic Research Associates	Con Edison/ PASNY	ff. Tr. 13,381

LIST OF WITNESSES - QUESTION 6

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Taylor Testimony	Testimony on the Economic Costs of Closing Indian Point, Testimony of Vince Taylor	Vince Taylor	Senior Economist for the Union of Concerned Scientists	UCS	ff. Tr. 13,298
Thornborough Testimony	Reorganized Testimony of John R. Thornborough	John R. Thornborough	Chairman of Westchester County Solid Waste, Recycling and Landfill Committee of the Environmental Management Council; member of Parents Concerned About Indian Point	Parents	ff. Tr. 14,380
Bohning Testimony	Testimony of Daryl Bohning	Daryl Bohning	Head of a Group of 10 Systems-Analysts and Analyst Programmers Setting Up a Comprehensive Data Processing System for the Lederle Laboratories Toxicology Facility	Parents	ff. Tr. 14,380
Wagers Testimony	Licensees' Testimony of William J. Wagers on Commission Question 6	William J. Wagers	Manager of Small Power Facilities Planning at Con Edison	Con Edison/ PASNY	ff. Tr. 13,701

LIST OF WITNESSES - QUESTION 6

Short Title	Full Title	Witness(es)	Position	Sponsor	Tr. Location
Wang Testimony	Licensees' Testimony of George C. S. Wang on Commission Question 6	George C. S. Wang	Forecast Development Manager in the Electric Forecast Section of the Forecasting and Economic Analysis Department at Con Edison	Con Edison/ PASNY	ff. Tr. 14,061

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

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

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)	:	
		July 11, 1983

CERTIFICATE OF SERVICE

I certify that I have served copies of CON EDISON'S PROPOSED OPINION, CONCLUSIONS OF LAW AND RECOMMENDATIONS AND PROPOSED FINDINGS OF FACT on the following by deposit in the United States mail, first class, this 11th day of July, 1983:

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U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

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Atomic Safety and Licensing Appeal Board Panel
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