UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

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

In the Matter of CONSOLIDATED EDISON COMPANY OF NEW YORK, INC. (Indian Point, Unit No. 2) POWER AUTHORITY OF THE STATE OF NEW YORK (Indian Point, Unit No. 3)	Docket Nos. 50-247 SP 50-286 SP January 24, 1983

LICENSEES' TESTIMONY OF STANLEY KAPLAN ON BOARD QUESTION 1.2

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I. Introduction

My name is Stanley Kaplan, Ph.D. I am an associate consultant at Pickard, Lowe and Garrick, Inc. My main area of work is in probability theory, risk and decision analysis, and particularly in probabilistic risk assessment methodology. I was a principal investigator on the Indian Point Probabilistic Safety Study. A statement of my professional qualifications is attached.

Board Question 1.2 asks:

> What bearing, if any, do the results reported in NUREG/CR-2497, "Precursors to Potential Severe Core Damage Accidents: 1969-1979, A Status Report" (1982), have upon the reliability of the IPPSS [Indian Point Probabilistic Safety Study]? 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?

II. Response

NUREG/CR-2497, "Precursors to Potential Severe Core Damage Accidents: 1969-1979, A Status Report" (1982) attempts to calculate the frequency of severe core damage (SCD) accidents in domestic nuclear power plants as a whole without any effort to distinguish between different plants or types of plants. It did this by sifting the Licensee Event Reports (LERs) for the period 1969-1979 and identifying incidents which it calls "precursors for potential severe core damage accidents." It then put these precursors through an event tree-type calculation procedure and derived an SCD frequency of 1.7 to 4.6 x 10^{-3} per year. This range of values is significantly higher than that calculated in most other probabilistic risk assessments (PRAs). For example, the Reactor Safety Study (RSS) calculated a core melt frequency of 5 x 10^{-5}, a factor of almost 100 less. The question, therefore, arises as to whether the NUREG invalidates the RSS, the process of PRA in general, and the IPPSS results in particular.

To address this question, it is helpful to paraphrase the methodology and line of argument of this NUREG. The essence of it is as follows:

> Up through 1979 we have had 432 years of reactor operation and one SCD accident, namely, TMI. We have also had a number of "near misses", e.g., Browns Ferry and Rancho Seco. We assign each of these near misses a "severity factor," which we get from the event trees. Adding these up, we consider that the near misses all together are the equivalent of about one more SCD accident. So we consider that the statistical experience, through 1979, is about two SCD events in 432 years which gives a frequency of 2/432 = 4.6 x 10^{-3}.

The way in which NUREG/CR-2497 evaluated those near misses can be and has been subject to much criticism both from an engineering modeling basis (*i.e.*, inaccuracies and oversimplifications in the event tree/severity factor work) and on the basis of statistical logic (*e.g.*, that near misses should not be counted at all). For example, with respect to the near miss contribution to SCD frequency, the Institute of Nuclear Power Operation (INPO) review of the NUREG (Ref. 1) concludes that the NUREG calculation of this contribution is about 30 times too high for the plants as they were at the time the events occurred. Furthermore, this review points out that the NUREG did not recognize the many improvements that have been made in light of the lessons learned from the events. Thus, INPO concludes that the NUREG frequencies are not appropriate either for past or future performance.

While I agree with much of this criticism, for our purposes here the evaluation of the near misses is not the main issue. For whether we consider that the experienced frequency of SCDs is one or two in 432 years makes little difference; both numbers are very different from the RSS results.

With regard to the RSS result, it should be noted that first counting all free world nuclear power reactors, there are today about 1,500 plant years of experience. Thus, our statistical evidence is now 1/1,500 rather than 1/432. Secondly, the RSS result is for a different event, core melt rather than severe core damage, in a different type of plant, Surry rather than TMI.

With respect to the PRA process itself, we should note that a PRA is, basically, a way of calculating the frequency of compound events from the frequencies of the "elemental" events which together make up the compound event. The way in which this is done is just pure logic. The NUREG does not in any way impugn or invalidate the basic process or methodology of PRA. It itself uses PRA methodology in its calculation of severity factors. Thus PRA, in general, in our view, is not called into question by the NUREG. However, in any particular application of PRA, there can, of course, be errors or omissions in logic or arithmetic. Thus, the Board's question appropriately asks specifically if the frequency of any IPPSS scenarios should be changed in light of the NUREG data.

The answer to this is that the data reported in the NUREG, and the incidents analyzed there were known to the IPPSS analysts at the time of their study. This knowledge was included in the scenario modeling and in the frequency calculations included in the IPPSS, along with the data from their own review of the LERs and, most importantly, along with the specific operating data from the Indian Point plants. The publication of the NUREG, thus, provided nothing basically new. It did, however, provide a useful focus of attention, particularly on the role of human errors of commission in the accidents of the past. It thus provided an opportunity to reexamine the IPPSS scenarios, particularly from the human error standpoint and from the standpoint of implementation of the lessons learned from past events. After this reexamination, the IPPSS analysts confirmed that the probability curves for the IPPSS scenario frequencies accurately express our state of knowledge of those frequencies, and that none of them require change in light of the NUREG. This statement is particularly true for the scenarios included in the IPPSS study which represent incidents of the TMI-type.

Further, it is worth noting that fundamental differences in the design of the Indian Point and TMI plants make it vastly less likely that scenarios of the TMI-type could occur at Indian Point. Among these design differences are 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. Even if the auxiliary feedwater were delayed as much as 20 to 30 minutes, the primary system would not experience a pressure transient, and the relie valve would not lift; therefore, it could not stick and could not lead to a loss of coolant.

If auxiliary feed were delayed longer than this, and if the relief valve were to lift and stick open, there is, at Indian Point, direct monitoring of the position of this valve so that the operators would be aware of its stuck 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 heat removal path required for this stability is provided by bleed and feed, forced circulation, and/or by natural circulation cooling which, at the Indian Point plants, is greatly enhanced by the elevated steam generator design.

One thing that has become clear through the post-RSS PRAs is that the frequency of core melt or damage can be very different from plant to plant because of design differences. For example, various PRAs of different nuclear power plants have reported core melt frequencies ranging from about 1.0×10^{-3} to 2.0×10^{-5} per reactor year (Ref. 2), a difference of about a factor of 50. Even within the same plant, the core melt frequency can change when there is a design modification. This was observed in the development of the IPPSS when such modifications were made to the Indian Point plants.

Finally, it should be recalled from the IPPSS study and the licensees' direct testimony on question 1 that a complete treatment of core melt frequency should be done with full probability curves rather than single number "point" estimates and that core melt frequency itself is a poor indicator of public health risks at Indian Point.

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2. Memorandum from William J. Dircks to NRC Commissioners (Jan. 5, 1983).

NAME

STANLEY KAPLAN

EDUCATION

Senior Post-Doctoral Fellowship, University of Southern California, 1967-1969.
Ph.D., Mechanical Engineering and Applied Mathematics, University of Pittsburgh, 1960. Post-doctoral courses in mathematics at the University of Pittsburgh and Carnegie Institute of Technology, 1960-1965.
M.S., Mechanical Engineering, University of Pittsburgh, 1958.
Graduate of the Oak Ridge School of Reactor Technology, 1955.
B.S., Civil Engineering, City College of New York, 1954.

PROFESSIONAL EXPERIENCE

General Summary

Mathematician and engineer well know for contributions to risk analysis and reliability theory, reactor physics, kinetics, and computational technique. Specializes in probabilistic methodology; decision theory; risk analysis; and, particularly, applications of Bayes' theorem. In this connection has worked specifically and recently on developing probabilistic and decision theoretic treatments of various phases of the energy business. Included here are PRA anlayses of several existing nuclear plants, hazardous material transportation and storage, spent fuel pools, aircraft impact, offshore oil drilling (environmental risk), underground oil storage, pipelines, and tarsands projects (business and construction risk). Developer of the DPD method for probabilistic calculations, the two-stage Bayesian technique for data analysis, the "set of triplets," "probability of frequency," "cause table," and "environmental table" concepts in risk analysis. Originator of the Matrix Theory of Event Trees and DPD approach to seismic risk analysis.

Chronological Summary

1977-Present	President, Kaplan & Associates, Inc., a consulting firm specializing in risk analysis and applied decision theory.
	Concurrently Adjunct Professor, Department of Chemical, Nuclear and Thermal Engineering, University of California, Los Angeles, and Associate Consultant, Pickard, Lowe and Garrick, Inc.
1975-1977	Private consultant specializing in risk analysis and decision theory.
1972-1975	Holmes & Narver, Inc., Anaheim, California. Director, Advanced Technology Division; Director, Systems Sciences Division; Technical Director, Nuclear & Systems Sciences Group.

1971-1972	Director of Software Development, COMARC Design Systems, Inc., San Francisco, California.
1969-1971	Product Manager and Senior Staff Member, Computer Sciences Corporation, Los Angeles, California.
1967-1969	Special Research Fellow, U.S. Public Health Service at University of Southern California, Los Angeles.
1955-1967	Westinghouse Bettis Atomic Power Laboratory, West Mifflin, Pennsylvania. Experimentalist, Experimentalist in Charge, Scientist, Senior Scientist, Fellow Scientist, Advisory Scientist.
1954	Lecturer, Department of Civil Engineering, City College of New York.
1962-1967	Concurrently Adjunct Professor of Mechanical Engineering, University of Pittsburgh; Lecturer, Department of Mathematics, Carnegie Institute of Technology.

MEMBERSHIPS

American Society of Civil Engineers.
American Nuclear Society.
Society of Industrial and Applied Mathematics.
New York Academy of Sciences.

REPORTS AND PUBLICATIONS

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