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)	)	June 20, 1983

LICENSEES' RESPONSE TO
REPORT OF PAUL J. AMICO

Consolidated Edison Company of New York, Inc. and the Power Authority of the State of New York, licensees of Indian Point Units 2 and 3, respectively, hereby respond to the Letter from Paul J. Amico to Administrative Judges (May 2, 1983) (Amico Letter), and to the Analysis of Probabilistic Risk Assessment Testimony: Indian Point ASLB Hearings Commission Question 1 (June 1, 1983) (Amico Analysis), as provided for by the Atomic Safety and Licensing Board's (Board's) Memorandum and Order of May 31, 1983.

The Board requested that Mr. Amico both summarize the issues pertaining to probabilistic risk assessment (PRA) and indicate whether additional testimony is needed for their

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resolution. Letter from James P. Gleason, Chairman, to Mr. Paul J. Amico (Mar. 7, 1983). Mr. Amico recommended additional testimony in eight areas. Amico Analysis § 6. These areas either (1) are sufficiently addressed in the record, (2) do not present issues which would materially affect the record, or (3) would require an unnecessary expenditure of time and resources.

Because this proceeding is already one year behind the schedule originally mandated by the Commission, and because reopening the record would only further delay the submission of the Board's recommendations to the Commission, additional testimony should be taken only if the Board is persuaded that such testimony is likely to have a material impact upon the validity or accuracy of the presentations of risk already contained in the record. The issue before the Board is not whether it possesses ultimate knowledge as to the risks posed by serious accidents at Indian Point, but whether the current state of knowledge indicates the risks of continued plant operation to be sufficiently large to justify plant shutdown. Licensees contend that the Commission was aware of this evolutionary state of PRA technology and accepted the use of models and techniques generally available at the start of the hearings as being sufficient to address the risk issue.

With one exception discussed below, it appears that receipt of further testimony would not have a material

impact upon the IPPSS results. Since the excepted matter can be best handled other than by reopening the record, the licensees oppose the receipt of additional testimony on the topics raised by Mr. Amico.

Intervenors concur, for different reasons, with licensees' position that the record should not be reopened. See Union of Concerned Scientists'/NYPIRG Response to Letter from P. J. Amico, May 2, 1983 (June 10, 1983) (UCS/NYPIRG Response); Letter from Richard M. Hartzman, Esq. to Administrative Judges (June 10, 1983).[1]

1. CRAC, CRAC2, and CRACIT

Mr. Amico first suggested that the Nuclear Regulatory Commission Staff "provide a detailed technical summary of the difference between the CRAC code used for their assessment and the improved version CRAC 2." Amico Analysis § 6, at 1. Differences between the original CRAC code, developed for use in the Reactor Safety Study (RSS), and CRAC2 are described in detail in the Sandia siting study, NUREG/CR-

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1. While intervenors suggest that licensees should not have "a second bite at the apple," UCS/NYPIRG Response at 2, licensees maintain that the record contains adequate support for the conclusions that the risk assessments performed for Indian Point were valid and comprehensive and that the risk of Indian Point is very low.

It should be further noted that Mr. Amico's report is not part of the record, was not submitted under oath, and has not been subject to cross-examination by any of the parties. Accordingly, it cannot serve as a basis for findings, conclusions, or recommendations in this proceeding.

2239, "Technical Guidance for Siting Criteria Development," (1982) (Appendix E).

Mr. Amico also suggested that the Staff "evaluate the potential effects on their final analysis of using CRAC 2 and should provide a reasonable justification of why they should not re-do their analysis using CRAC 2, since it is supposedly a more accurate code than CRAC, or else should provide the results of a re-analy[s]is using CRAC 2." Amico Analysis § 6, at 1. Further analysis will not provide the Board with new information to aid them in rendering a judgment because (1) differences between CRAC and CRAC2 are already described in documents available to the Board, (2) Staff has already provided their reasons for using CRAC, (3) the codes have been demonstrated to be virtually identical in results, with any differences due to input assumptions rather than the codes, and (4) licensees presented testimony based upon the most accurate code available.

CRAC2 provides a more realistic model of several parameters of potential importance in consequence assessment than does CRAC. However, Staff does not "anticipate that we would see significant differences at all between the CRAC and CRAC-II [sic] for these particular calculations that have been made." Tr. at 8622 (Blond). With the sole exception of emergency response modeling and assumptions, benchmark tests designed to evaluate the differences in these

codes confirm Staff's position.[1] Furthermore, Staff modified CRAC in assessing the Indian Point risk to update important evacuation response assumptions and models. See Joint Testimony of Sarbeswar Acharya and Roger M. Blond Regarding Calculational Methodology of the CRAC Code at III.C.B-4 to III.C.B-6.

Additionally, the CRACIT code used in licensees' analysis does incorporate the more accurate parameter modeling included in CRAC2, and allows long duration releases to be modeled as a series of "puffs," which Staff agrees is a more realistic model. Tr. at 8647 (Blond). It is, thus, the most accurate and realistic consequence model presented at the proceeding.

Mr. Amico acknowledges that "the real significant contributor" to Staff's and licensees' consequence results is the emergency response assumptions used by Staff for regional disasters, not the details of the computer codes. In fact, Mr. Amico states that "this issue is the only significant issue in the ex-plant consequence analysis." Amico Analysis § 4, at 14 (emphasis added).

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1. D.C. Aldrich, et al., International Standard Problem for Consequence Modeling: Results, in Proceedings of the International ANS/ENS Topical Meeting on Probabilistic Risk Assessment at 787 (Sept. 20-24, 1981).

2. MARCH 1.1 and MARCH 2.0

Mr. Amico recommended consideration of the MARCH 2.0 code rather than the MARCH 1.1 version for the containment analysis. Amico Analysis § 6, at 1. Both Staff and licensees used versions of MARCH in their analyses of serious accidents. Because both parties are aware of limitations in this code, their assessments were supported by additional analyses, including the use of supplemental computer codes. See Indian Point Probabilisitc Safety Study (IPPSS) at 4.1-1 to 4.1-13; NUREG-0850, "Preliminary Assessment of Core Melt Accidents at the Zion and Indian Point Nuclear Power Plants and Strategies for Mitigating Their Effects" at 3-1, Vol. 1 (1981). While an improved computer code would update the current model, thereby limiting the additional analyses now required of containment analysts, additional testimony at this time based upon a code still undergoing peer review will not clarify or improve the record which is before the Board.

Mr. Amico also questions the use of the CORRAL 2 code as opposed to NAUA 4 in calculating radioactive releases. Amico Analysis § 6, at 1-2. Unlike NAUA, CORRAL does not fully consider natural fission product removal processes such as agglomeration and steam condensation on aerosols. Thus, the Indian Point release fractions as computed in the IPPSS and by the Staff are conservatively overstated. This leads to a substantial overstatement of risk in both IPPSS

and Staff analyses, particularly for early fatalities. See Testimony of Dr. Sarbeswar Acharya Regarding NRC Staff Assessment of Accident Consequences and Risks at III.C.A-20 to III.C.A-21 (Acharya Testimony).

Inasmuch as the analytical tools and models used in PRA are currently undergoing continuous development and refinement, Mr. Amico's reasoning could be used to defer any Board decision indefinitely because there undoubtedly will be further revisions on these models in the future beyond those identified by Mr. Amico.

3. Source Term Developments

The licensees agree with Mr. Amico that the present Commission and industry (IDCOR),[1] research regarding source terms is important, and that "[i]f this work shows a source term reduction on the order of a factor of ten, as is the personal belief of Witness [Robert] Bernero, this will cause a substantial impact on the results." Amico Letter at 2; see Amico Analysis § 6, at 2. Sandia National Laboratories performed sensitivity studies to evaluate the impact of source term reductions upon the potential consequences of reactor accidents. They concluded that "[m]ean early fatalities . . . are decreased dramatically (about two orders-of-

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1. IDCOR is the Industry Degraded Core Rulemaking.

magnitude) by a one order-of-magnitude decrease in source term SST1."[1] NUREG/CR-2239 at 1-3.

The NAUA 4 and CORRAL differences discussed above are but a part of larger fission product behavior issues. Licensees presented the testimony of two noted experts in the field of radiation chemistry, Dr. William Stratton and Dr. Walton Rodger,[2] regarding detailed, plant-specific calculations of fission product behavior for the two most important accident sequences at Indian Point. See Licensees' Testimony of William R. Stratton, Walton A. Rodger, and Thomas E. Potter on Question One at 68-86. Based upon the more realistic source term calculations of Drs. Stratton and Rodger, a smaller Emergency Planning Zone may be warranted. Id. at 6. Staff testified that these calculations were "reasonable," and that they had neither data nor information indicating that the calculations were wrong. Tr. at 12,585, 12,589 (Bernero).

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1. This source term would be associated with a relatively rapid release of radioactivity.

2. Dr. Stratton, a member of the Advisory Committee on Reactor Safeguards from 1966 to 1975, was also on the staff reporting to the President's Commission on the Accident at Three Mile Island. Dr. Rodger is the lead contractor for IDCOR's section on Fission Product Transport and is the 1981 recipient of the American Institute of Chemical Engineers Robert E. Wilson Award.

While the use of more realistic source terms will have a major impact upon certain presentations of risk in the record, the Stratton, et al. testimony provides a sufficient basis for the Board to consider fission product behavior issues in making its findings on the Commission's Questions as of the present time. Awaiting further Commission and industry research would unduly prolong this proceeding.

Additionally, there is no reason to delay, as Mr. Amico suggests, a final decision based upon the results of ongoing research because, notwithstanding the current overestimation of the source term in the IPPSS and Staff presentations, the risk of the plants is already demonstrated in the record to be extremely low. This conclusion is reached by both Staff and licensees, despite differences in source term calculations. More realistic source terms would only reinforce that conclusion.

4. Stress and Aging Factors

Failure rates used in the IPPSS appropriately account for aging and stress effects. Because failure rates are derived from data on a broad mix of equipment of various ages both at Indian Point and other nuclear power plants, any failures due to aging are embedded in these failure rates. Tr. at 7380-82 (Bley). Stress effects are explicitly considered in determining failure/success probabilities. The use of a constant factor, as raised by Mr. Amico, would be an oversimplification of these effects. See

Tr. at 7033-35 (Shon, Kaplan, Bley). Therefore, the risk analysis presented to the Board provides an adequate treatment of aging and stress effects upon nuclear power plant equipment, and the receipt of further evidence on this topic would not have a material impact on that analysis.

5. Sabotage

Mr. Amico also proposed that the Board receive "[e]xpert testimony on the significance of sabotage." Amico Analysis § 6, at 2. In its "Policy Statement on Safety Goals for the Operation of Nuclear Power Plants," the Commission stated, with respect to the possible effect of sabotage or diversion of nuclear materials, that "[a]t present there is no basis on which to provide a measure of risk on these matters. It is the Commission's intention that everything that is needed shall be done to keep such risks at their present, very low, level; and it is our expectation that efforts on this point will continue to be successful." 48 Fed. Reg. 10,772, 10,773 (Mar. 14, 1983). Thus the Commission has, as a matter of policy, acknowledged that explicit treatment of sabotage in risk quantification studies is presently beyond the state-of-the-art, but that present regulatory requirements, such as those employed at Indian Point, are sufficient to keep such risks acceptably low. These requirements are included in extensive and

detailed Commission regulations and procedures which adequately address the issue of nuclear power plant security.[1]

Sabotage was specifically excluded from the IPPSS because the project team determined that it was not appropriate in a study that was to be made public. Tr. 7042, 7050, 7147 (Garrick). As Mr. Amico recognized, Amico Analysis § 6, at 2, it would obviously be inadvisable to present testimony on sabotage and security issues specific to Indian Point.

Thus, testimony in this area would not aid the Board in assessing the level of risk at these plants. Licensees continue to believe that testimony such as that suggested by Mr. Amico would be inappropriate in a public forum, and should certainly not be explored initially by the Commission on other than a generic or rulemaking basis.

6. Point Estimates

Mr. Amico suggested that the licensees "present the level 1 (point estimate) risk curves for external initiators and total risk rather than just the level 2 (confidence

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1. See, e.g., 10 C.F.R. Part 34(c) (physical security plan required); 10 C.F.R. Part 73 (physical protection of plants and materials); Reg. Guide 5.7 (Entry/Exit Control for Protected Areas, Vital Areas and Material Access Area); Reg. Guide 5.14 (Use of Observation (visual surveillance) Technique in Material Access Areas); Reg. Guide 5.43 (Plant Security Force Duties); Reg. Guide 5.44 (Perimeter Intrusion Alarm System); Reg. Guide 5.54 (Standard Format and Content of Safeguards Contingency Plans for Nuclear Power Plants); NUREG 0464, "Site Security Personnel Training Manual."

level) curves for these items." Amico Analysis § 6, at 2. In presenting Level 2 risk curves, which include an explicit quantification of uncertainty, licensees have submitted to the Board a complete picture of the risk at Indian Point. By calculating uncertainty in a manner that is at the "forefront" of the methodology, Tr. at 8614 (Blond), the authors of IPPSS have been responsive to the Lewis Committee and other reviewers of the RSS who advocated a frank presentation of uncertainty. Submitting point estimates, which have no precise definitions, see Tr. at 8247 (Kaplan), would not contribute to the Board's recommendation to the Commission regarding "[w]hat risk may be posed by serious accidents at Indian Point 2 and 3." In re Consolidated Edison Company of New York, Inc., 14 N.R.C. 610, 612 (Sept. 18, 1981). While it may not be "appropriate" to compare point estimates provided by Staff with risk curves provided by licensees, see Acharya Testimony at III.C.A-39 n.*, III.C.A-43 to III.C.A-45, such a comparison is not necessary to answer the Commission's question.

Staff has presented its assessment of the health risk at Indian Point, based on point estimates, and has concluded that the risk is not high, Direct Testimony of Frank H. Rowsome to Contention 1.1 and Board Question 1.1, at 2, that it is within the range of risks posed by other nuclear power plants, and that it meets the Commission's preliminary safety goals. Direct Testimony of Frank Rowsome and Roger

Blond Concerning Commission Question 5, at [A]33, B-3, B-13, B-16. Licensees presented a more complete analysis of the risk and arrived at the same conclusions. The record is complete regarding the level of risk posed by serious accidents at Indian Point.

7. Safety Goal Comparisons

The information is already available for the direct comparison recommended by Mr. Amico, Amico Analysis § 6, at 3, by the Staff of their results with the Commission's preliminary safety goals. Staff estimates of core melt frequencies are in Table 1 of the Staff's testimony on damage state likelihoods. Direct Testimony of Frank H. Rowsome Concerning Damage State Likelihoods at Table 1 (row labeled "grand total"). Staff estimates of societal risk within 50 miles have also been presented, see Direct Testimony of Frank H. Rowsome to Contention 1.1 and Board Question 1.1, at 2-3; Testimony of Dr. Sarbeswar Acharya Regarding NRC Staff Assessment of Accident Consequences and Risks at Tables III.C-8 and III.C-9, as have estimates of risk to individuals within one mile of the plant with supportive medical treatment. Testimony of Dr. Sarbeswar Acharya Regarding NRC Staff Assessment of Accident Consequences and Risks at Figure III.C.50.

8. Pressurized Thermal Shock

The Board has received sufficient testimony regarding the importance of pressurized thermal shock (PTS) to the

risk at Indian Point. Licensees presented the low frequency of vessel failure from all causes, of which PTS is only a subset. They also presented a conservative approximation of the probability of extension without arrest of cracks in the vessel. See Licensees' Testimony of Dennis C. Richardson and Dennis C. Bley on Board Question 1.4, at 2, 3. Because the Indian Point plants have not reached a level in which the Commission's screening criterion for acceptability regarding PTS would suggest the problem even needs further analysis, Staff concluded that the "present and continuing acceptability of PTS risk is assured." Dr. Hugh W. Woods and Raymond W. Klecker on Board Question 1.4, at 10. Additional testimony on this issue will not add to the Board's state of knowledge concerning the risk at Indian Point, nor materially affect presentation of that risk.

Conclusion

For the foregoing reasons, the Board should not delay the completion of this proceeding by reopening the record.

Respectfully submitted,

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NUCLEAR REGULATORY COMMISSION

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CERTIFICATE OF SERVICE

I hereby certify that on the 20th day of June, 1983, I caused a copy of Licensees' Response to Report of Paul J. Amico, to be served by first class mail, postage prepaid on the following:

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