ATTACHMENT 2

SHORT TERM RISK ESTIMATES FOR INDIAN POINT UNITS 2 AND 3

May, 1980

Power Authority of the State of New York
Indian Point Unit No. 3
Docket No. 50-286
Facility Operating License No. DPR-59

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
Facility Operating License No. DPR-26

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Short Term Risk Estimates
for Indian Point Units 2 and 3

This study extends the risk perspective presented in Reference 1 to include the risk from short-term health effects which could result from a severe accident at Indian Point. These risks were calculated using the computer code CRAC, which was developed as part of the Reactor Safety Study, Reference 2.

Two of the fundamental inputs to the CRAC code are the estimated quantity of fission products released to the atmosphere in the various accident sequences and the associated accident sequence probabilities. In WASH 1400, the spectrum of possible fission product releases (associated with the various combinations of severe accident sequence and containment failure mode) was divided into seven discrete categories. Each combination of accident sequence and containment failure mode was then assigned to the most appropriate release category. The same approach was used in this study. The probability for each release category was obtained by summing the probabilities of the assigned accident sequences and then smoothing between adjacent release categories[(1)]. Smoothed probabilities for each release category were used as input to CRAC. The basic accident sequence probability estimates (before smoothing) were obtained from Reference 1.

CRAC also requires as input site meteorological and demographic data and core fission product inventories. Demographic and meteorological data specific to the Indian Point site were utilized. The core fission product inventory used in the Reactor Safety Study calculations was adjusted in proportion to the power output from the Indian Point plants for use in these calculations. The same evacuation model employed in the Reactor Safety Study was used here and

(1) Smoothing was accomplished in the same manner as in WASH 1400 in order that risks calculated in this study might be compared with WASH 1400 results on a consistent basis. The basic concept of smoothing, which is used to account for uncertainty in release magnitude, is described on page V-33 of Appendix V, of Reference 2. The licensees do not endorse the use of smoothing and in fact agree with the Risk Assessment Review Group, Reference 4, that smoothing tends incorrectly to increase the calculated levels of risk. As noted earlier, smoothing is used here for the sole purpose of consistent risk comparison.

provides a common frame of reference. Development of a more sophisticated evacuation model for the Indian Point site is underway as part of future work but was not feasible in the time available for this study.

Risk characteristics were calculated for the Indian Point units for the following cases:

Case 1: The plants as they were designed and operated at the end of 1979. The probabilities per release category were as determined for Case 2 in Reference 1. Compared to WASH 1400, the probability of steam explosion was assumed to be 1-to-2 decades lower and, for the LOCA sequences, the probability of basemat melt-through was also assumed to be 1-to-2 decades lower.

Case 2: The plants as modified by the NRC Interim Order of February 1980, reference (3). Accident sequence probabilities were reduced from those estimated in Reference 1 in order to estimate the potential risk reduction from the increased testing of interfacing check valves and for the increased operator training required in the NRC Order.

Study results are plotted in Figure 1 in the form of complementary - cumulative distribution functions. In addition to the results for Cases 1 and 2, described above, Figure 1 contains two additional risk characteristic curves plotted directly from WASH 1400 results. The curves, labeled Cases A and B are the following:

Case A: The risk characteristic curve for the reference PWR, as analyzed in WASH 1400, located at a hypothetical site having characteristics defined in WASH 1400 as the composite of 68 actual sites.

Case B: The risk characteristic curve for the reference PWR, as analyzed in WASH 1400, located at the Indian Point site. This is the case relied upon by the NRC Staff in their presentation to the NRC Commissioners wherein it was suggested that the Indian Point plants represent a higher level of risk than do reactors at other sites. This comparison was inappropriate because the specific Indian Point plant designs were not taken into account.

Figure 1 clearly shows that risk for the Indian Point plants (before any mitigation actions) falls below the composite risk curve of WASH 1400 (Case A, defined above). This composite risk characteristic represents a level of risk which has implicitly been judged acceptable in past licensing actions. The effect of the actions already agreed to (Case 2, as defined above) should be a substantial reduction to the already moderate risk levels. In fact the risk characteristic for Case 2 lies totally below the origin probability value of 10^{-9} per reactor year and is therefore indicated only by a note on Figure 1. From Figure 1 it is apparent that the early NRC estimates of risk from the Indian Points plants are too high by a substantial amount. This resulted from considering only the specific site characteristics and not the specific plant design features as well[2].

To assess the effect of assuming a low probability for steam explosion and basemat melt-through, risk characteristics were also calculated for the probabilities of steam explosion and basemat melt-through assumed in WASH 1400 (0.01 for steam explosion and generally 0.9 for basemat melt-through). The calculated results (which have not been shown on Figure 1) differ from Case 1 by an insignificant amount, and show that the short-term risk characteristics for Indian Point are not sensitive to the probability assumed for containment failures by steam explosion or basemat melt-through.

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[2] In his letter to each of the licensees dated April 9, 1980, Mr. Harold R. Denton identified Enclosure (5) (to his letter) as containing "information concerning the contribution to societal risk presented by the Zion and Indian Point Units, that was used as background material for the staff briefing of the Commissioners on February 5, 1980." Enclosure (5) is a draft paper by R. Blond wherein the following statement appears at page 4, "Once again it should be pointed out that these curves assume a Surry type PWR design at Indian Point and Zion. To perform the analysis properly, the specific systems interactions for the Indian Point and Zion designs should be factored into the problem." (Emphasis added).

Conclusions

The principal conclusion to be drawn from this study and from Reference 1 is that the level of risk associated with the Indian Point plants is less than the level of risk which has found implicit acceptance in past licensing actions, i.e., the level of risk reported in WASH 1400 for the reference PWR located at an average or "composite" site. This conclusion results from accident probability estimates based largely on the application of WASH 1400 methods and data to the specific design of Indian Point plants. It is noted that the use of WASH 1400 for comparative purposes was both endorsed and encouraged in Reference 4 by the WASH 1400 Risk Assessment Review Group. A second conclusion of importance is that consideration of both the site characteristics (demography and meteorology, etc.) and the specific plant design are essential before conclusions are drawn concerning the risk from severe accidents for a particular reactor at a particular site.

References

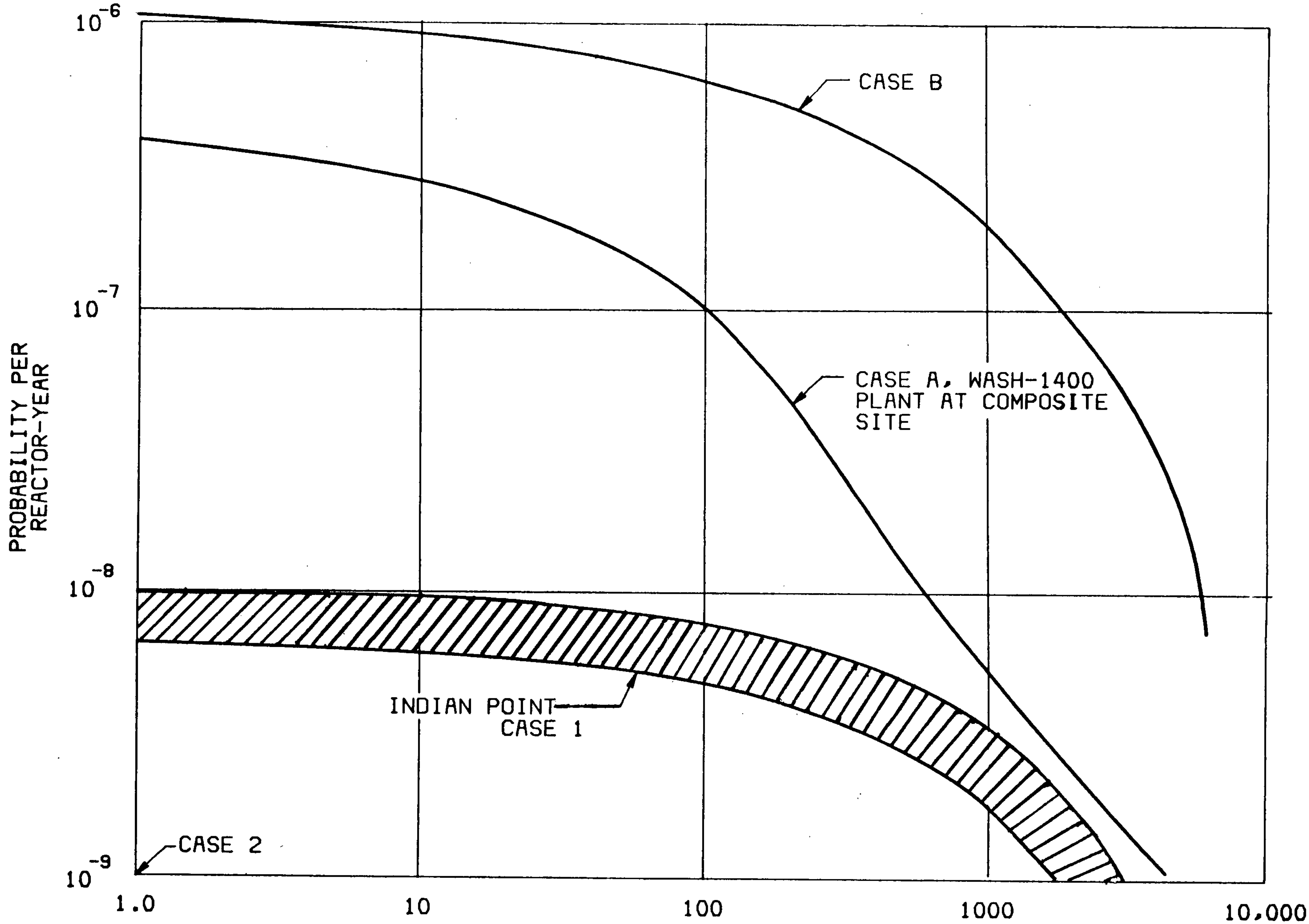
1) "An Evaluation of Residual Risks from the Indian Point Nuclear Power Plants", Offshore Power Systems Report 35A96, May 1980.

2) "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants", WASH 1400, October 1979.

3) Nuclear Regulatory Commission Orders in the matter of Consolidated Edison Company (Indian Point Station, Unit No. 2), Docket No. 50-247, and in the matter of Power Authority of the state of New York, (Indian Point Station, Unit No. 3), docket No. 50-286, February 11, 1980.

4) H.W. Lewis, et.al, "Risk Assessment Review Group Report to the U.S. Nuclear Regulatory Commission" NUREG/CR-0400, September 1978.

FIGURE 1



10^{-6}
10^{-7}
10^{-8}
10^{-9}
PROBABILITY PER REACTOR-YEAR
CASE B
CASE A, WASH-1400 PLANT AT COMPOSITE SITE
INDIAN POINT CASE 1
CASE 2
1.0
10
100
1000
10,000

CONSOLIDATED EDISON COMPANY of NEW YORK, INC.
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POWER AUTHORITY of the STATE of NEW YORK
10 COLUMBUS CIRCLE, NEW YORK, N.Y. 10019

May 23, 1980

Re: Indian Point Unit No. 2
Docket No. 50-247

Indian Point Unit No.3
Docket No. 50-286

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

SUBJECT: INDIAN POINT PLANT-SPECIFIC RISK EVALUATION

Dear Mr. Denton:

The purpose of this letter is three-fold: first, to provide the NRC with the preliminary results of studies that estimate the residual risk connected with operation of Indian Point Units 2 and 3; second, to advise you that the characterization of this risk by the NRC staff is inconsistent with the conclusions of our studies; and third, to urge that any further NRC staff conclusions with respect to operation of Indian Point Units 2 & 3 be based on plant-specific residual risk studies.

In a March 5, 1980 presentation to the ACRS, and in previous presentations to the Commissioners, the Zion and Indian Point plants have been characterized by the NRC staff as comprising more than 30 percent of the national risk from nuclear reactors. This characterization appears to be the motivating force in the Zion/Indian Point Near Site Studies, and also the consideration of adjudicatory hearings respecting Indian Point. The Consolidated Edison Company of New York, Inc. and the Power Authority of the State of New York have jointly concluded that this characterization is erroneous, and the risk posed by the Indian Point units greatly exaggerated by the NRC staff.

Your letter of April 9, 1980 included as Enclosure 5 a NRC memorandum from Mr. Mat Taylor of the Probabilistic Analysis Staff to Mr. Frank Rowsome, Deputy Director Probabilistic Analysis Staff. This NRC memorandum and its attachments detail the analysis and assumptions used by the staff to support your characterization of Indian Point. On page 4 of the attachment to the NRC memorandum is the following statement,

> "Once again it should be pointed out that these curves assume a Surry type PWR design at Indian Point and Zion. To perform the analysis properly, the specific systems interactions for the Indian Point and Zion designs should be factored into the problem. However, for an initial cut, it is not anticipated that the design differences would substantially change the results."

The Consolidated Edison Co. of New York, Inc. and the Power Authority of the State of New York have performed the analysis recommended by the NRC memorandum, with provision for the specific systems interactions of the Indian Point design. We have concluded that the anticipations of the NRC staff are not applicable to our plants, and that an appropriate plant-specific analysis such as we have performed results in a totally different characterization.

Attachment 1 to this letter is a Westinghouse/Offshore Power System (OPS) Report on the Evaluation of Residual Risk for the Indian Point Power Plant. Attachment 2 is a report by Dr. Ian Wall of the Electric Power Research Insititute, discussing his inclusion of the plant-specific probabilities from the OPS Report in an analysis similar to the one performed by the NRC staff. The objective of these studies, reported in Attachments 1 and 2, was to establish within a short period of time a reasonable estimate of the residual risk for the Indian Point Nuclear Station Units 2 and 3. In order to present a frame of reference for comparative purposes, these studies are based on the general methodology and data of the Reactor Safety Study, WASH 1400.

These results, presented as risk curves, were initially provided to the NRC staff in an oral presentation on February 20, 1980 and again by docketed letter on February 25, 1980.

The risk curves resulting from the attached studies show that the risk from short term effects at the Indian Point site falls significantly below the WASH 1400 risk curve for a PWR at a composite site (which is the average of 68 actual sites). This is to be expected because of the special design features installed as a result of the original licensing review on these plants, and indeed you recognized this factor in a written decision to the Commission dated February 11, 1980, where you observed that these special design features "would limit the potential radiological consequences of a major accident." We find that the 30 percent risk figure as set forth in Mr. Taylor's memorandum above, is inconsistant with your conclusion. In contrast to an analysis of the actual situation, when a Surry-type PWR is hypothetically placed at the Indian Point site, the resulting calculated risk is greater than the composite curve by up to a factor of 10. This latter comparison was the basis for the NRC staff presentation to the Commissioners suggesting that the core melt risk at the Indian Point site may be unacceptably high and should be reduced.

The principal conclusion to be drawn from the attached studies is that the level of risk associated with the Indian Point plants is significantly less than the level of risk which has found implicit acceptance in past NRC licensing actions, the level of risk reported in WASH 1400 for a typical PWR located at an average or "composite" site. This conclusion results from accident probability estimates based largely on the application of WASH 1400 methods and data to the specific design of the Indian Point plants. It should be noted that the use of WASH 1400 for such comparative purposes was both endorsed and encouraged by the WASH 1400 Lewis Review Panel.

A second conclusion of importance is that consideration of both the site specific characteristics (demography, meteorology, etc.) and the plant-specific design are essential before responsible conclusions may be drawn concerning the risk from core melt accidents for a particular reactor at a particular site. It has been apparent at meetings with your staff and their consultants that consideration of plant specific probability and quantitative risk assessment have been excluded from their scope of review. Considering the extent of the Commission's interest in reactor risk data, as well as the effort and the millions of dollars which are being expended on these studies by both the utilities and the NRC, any reliance upon risk assessment studies for predictive or regulatory purposes should be based on a more complete and comprehensive analysis, including in particular a WASH 1400 plant-specific quantitative study. In view of the erroneous conclusion of the staff's "initial cut" evaluation as demonstrated by our own plant-specific evaluation, we believe that a plant-specific evaluation should be performed by the NRC before any determinations are made.

In view of the deliberations by the Commissioners in regard to interim operation of Indian Point and their apparent conclusions as to a need for an adjudicatory hearing, we submit the attached reports for our dockets. We believe the attached reports contain significant information which should be bought to the Commissioner's attention to assist the deliberations currently underway. Copies of this letter and its Attachments have been sent directly to the Commissioners for their information.

Should the Commissioners or you or Staff have any questions, please contact us.

Very truly yours,

Paul J. Early
Vice President and
Assist. Chief Engineer
(Projects)
Power Authority of the
State of New York

William J. Cahill, Jr.
Vice President
Consolidated Edison Co.
of New York, Inc.

Attachment

cc: Peter Crane, Esq., NRC General Counsel's Office
U.S. Nuclear Regulatory Commission
John F. Ahearne, Chairman
Peter A. Bradford, Commissioner
Victor Gilinsky, Commissioner
Joseph M. Hendrie, Commissioner
Richard T. Kennedy, Commissioner
Samuel J. Chilk, Secretary to the Commission
Ellen Weiss, Esq.