

William J. Cahill, Jr.
Vice President

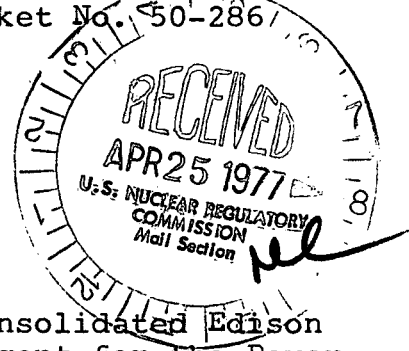
Regulatory

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Consolidated Edison Company of New York, Inc.
4 Irving Place, New York, N Y 10003
Telephone (212) 460-3819

Re: April 20, 1977
Indian Point Unit No. 3
Docket No. 50-286

Mr. Ben Rusche, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555



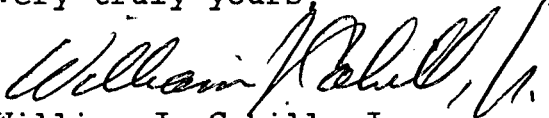
Dear Mr. Rusche:

By letter dated January 26, 1977, Consolidated Edison Company, Inc. on its own behalf and as agent for the Power Authority of the State of New York submitted a document entitled, "Analysis Of The Emergency Core Cooling System In Accordance With The Acceptance Criteria Of 10CFR50.46 And Appendix K of 10CFR50." As requested by the Regulatory Staff subsequent to that submittal, additional information, demonstrating that the $C_D = 1.0$ case is the most limiting case for Indian Point Unit No. 3, is provided herewith.

The Indian Point Unit No. 2 analysis demonstrated the limiting break to be the $C_D = 1.0$, DECLG case. The break spectrum results for Indian Point Unit No. 2 are applicable to demonstrating that the $C_D = 1.0$ case is limiting for Indian Point Unit No. 3. However, to provide further justification the Indian Point Unit No. 3 reanalysis was also carried out for the $C_D = 0.8$ case. In addition, it should be noted the reanalysis assumed an upper head temperature equal to the hot leg fluid temperature, and the results reported in WCAP-8855 demonstrate the limiting break is unchanged for this assumption.

A more complete discussion is contained in Attachment 1.

Very truly yours,


William J. Cahill, Jr.
Vice President

WJC:nvg

Copy to: Mr. George T. Berry
General Manager and Chief Engineer
Power Authority of the State of New York
10 Columbus Circle
New York, N.Y. 10019

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INDIAN POINT UNIT NO. 3 ECCS ANALYSIS

Indian Point Units No. 2 and 3 differ from the generic 4 loop 15X15 plant analyzed in WCAP-8855 in accumulator water volume and total accumulator volume. These differences have a substantial effect on core cooling during the refill and reflood transient. Indian Point Unit No. 2 and 3 analyses, using both the March and October 1975 evaluation models, demonstrate only a partially filled downcomer at the time of accumulator emptying. The downcomer water level varies with break size with the $C_D=1.0$ case having the lowest water level. The water level increases with decreasing break size (i.e. decreasing C_D). This variation is significant, and results in the $C_D=1.0$ case yielding the greatest peak clad temperature (PCT). By comparison the 1400 cu. ft. total accumulator volume used in the generic 4-loop 15X15 plant is sufficiently large to provide a fully filled downcomer at the time of accumulator emptying for the entire spectrum of break sizes, including the $C_D=1.0$ case. As a result the maximum peak clad temperature does not occur in the $C_D=1.0$ case for the generic analysis.

The Indian Point Units No. 2 and 3 analyses are identical in the major parameters, including:

Blowdown and Reflood Transient Core Power - 102% of 3220MWT

Total Accumulator Volume - 1100 cu. ft.

Accumulator Water Volume - 800 cu. ft. (October 1975 evaluation model analyses of Indian Point Units No. 2 and 3)

Steam Generator Tube Plugging - 0% (Reflood and blowdown transient for the March 1975 evaluation model analyses of Indian Point Units No. 2 and 3)

Steam Generator Tube Plugging - 4% (Blowdown transient for October 1975 evaluation model analyses of Indian Point Units No. 2 and 3)

Minor differences in the analyses are:

Accumulator water volume - 770 cu. ft. for Indian Point Unit No. 2 March evaluation model analysis versus 800 cu. ft. for Indian Point Unit No. 3 March evaluation model analysis. This difference is not significant as the downcomer level, at the end of accumulator injection, is only slightly affected and the trend with break size is unchanged.

Steam generator tube plugging - 4% for the reflood transient for Indian Point Unit No. 2 versus 0% for the Indian Point No. 3 analysis - both for the October 1975 evaluation model analyses. Comparison of the core reflood parameters demonstrates this change to be negligible, and 4% steam generator tube plugging does not shift the worst break.

Peak rod kw/ft - 102% of 13.4 kw/ft for Indian Point Unit No. 2 (equivalent to 102% of 2758 MWT) and 102% of 14.5 kw/ft for Indian Point Unit No. 3 (equivalent to 102% of 3025 MWT). This difference has no effect on the blowdown or reflood system transient.

Based on the similarities of these analyses the Indian Point Unit No. 3 analysis can reference the Indian Point Unit No. 2 March 1975 evaluation model analysis as the basis for the $C_D = 1.0$ as the case yielding the maximum PCT. (Previous Indian Point Unit No. 3 submittal utilized a mixture of the December 1974 and March 1975 evaluation models - and the differences between these models make this break spectrum inappropriate for comparison.)

Additional confirmation is provided by considering trends in PCT versus break size for Indian Point Unit No. 2 (March and October evaluation models) and the Indian Point Unit No. 3 analysis (October model). For Indian Point Unit No. 2 the drop in PCT, going from $C_D = 1.0$ to 0.8, was 30 and 37°F, for the March and October evaluation models respectively. For Indian Point Unit No. 3 the drop was 50°F, thus further demonstrating the Indian Point Unit No. 3 limiting break is the $C_D = 1.0$ case.

William J. Cahill, Jr.
Vice President

REGULATORY DOCKET FILE COPY

Consolidated Edison Company of New York, Inc.
4 Irving Place, New York, N Y 10003
Telephone (212) 460-3819

January 26, 1977
Indian Point Unit No. 3
NRC Docket No. 50-286



Mr. Ben C. Rusche
Director
Office of Nuclear Reactor
Regulation
U.S. Nuclear Regulatory
Commission
Washington, D.C. 20555

Dear Mr. Rusche:

On September 9, 1976 the Commission published in the Federal Register an Order for Modification of License issued on August 27, 1976 with respect to Indian Point Station, Unit No. 3. 41 Fed. Reg. 38227 (1976). That Order modified Facility Operating License No. DPR-64 by adding the following new provisions:

1. As soon as possible, the Licensee shall submit a reevaluation of ECCS cooling performance calculated in accordance with an approved Westinghouse Evaluation Model, with appropriate correction for upper head water temperature.
2. Until further authorization by the Commission, the Technical Specification limit for total nuclear peaking factor (F_0) shall be reduced to 2.31.

In accordance with and in satisfaction of the first of these provisions, Consolidated Edison Company, Inc. on its own behalf and as agent for the Power Authority of the State of New York, as coholders of Facility Operating License No. DPR-64, has consulted with Westinghouse, the nuclear steam system supplier, and is pleased to submit herewith its reanalysis of the Indian Point 3 Emergency Core Cooling System. This reanalysis has been performed in accordance with the October 1975 Westinghouse Evaluation Model. This Model has been approved by the Regulatory Staff, and the analysis incorporates, as required, a correction for upper head water

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Mr. Ben C. Rusche

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January 26, 1977

temperature applicable to this facility. Based on the results of this reanalysis the peaking factor should now be returned to its original level of 2.32.

We and our vendor are available to meet with you or members of the Staff, if necessary, to review this reanalysis. Upon acceptance by the Staff, we assume that appropriate action will be taken to vacate the special license provisions quoted above.

In accordance with Section 2.708 (d) of the Commission's Rules of Practice, we are submitting one (1) original and twenty (20) copies of this letter and its enclosure.

Very truly yours,



William J. Cahill, Jr.
Vice President

Enclosure

CC: Mr. George T. Berry
General Manager and Chief Engineer
Power Authority of the State of
New York
10 Columbus Circle
New York, New York 10019

Mayor George V. Begany

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Secretary, USNRC (3)