William J. Cahill, Jr.
Vice President

Regulatory Docket File

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

RECEIVED JUL 24 1975 U.S. NUCLEAR REGULATORY COMMISSION Mail Section

July 23, 1975

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

Mr. George Lear, Chief
Operating Reactors Branch #3
Division of Reactor Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

DOCKETED USNRC JUL 24 1975 MAIL SECTION DOCKET CLERK

DL

Dear Mr. Lear

In response to your letter of July 16, 1975, we are attaching (Attachment A) to this letter the information you requested regarding our proposed spent fuel pool modification. We are also attaching (Attachment B) documentation of information requested by your Staff by telephone on June 10, 1975 concerning structural aspects of the proposed modification.

Regarding your June 12, 1975 letter, the Consolidated Edison Nuclear Facilities Safety Committee has reviewed the planned modification and concurs that the change has no adverse effect on nuclear safety. However, your request for information has indicated that changes to the Technical Specifications are necessary for the specifications to be consistent with the modified fuel storage facility; therefore, Commission review and approval is needed prior to implementation of the planned modification.

We appreciate the timely review the Commission is providing, and we will be pleased to supply any additional information that might be required by you and your Staff in conjunction with this review.

Very truly yours

William J. Cahill, Jr.

William J. Cahill, Jr.
Vice President

enc.
mk

7885

Attachment A

Additional Information Requested by U.S.N.R.C. Letter dated July 16, 1975
Indian Point Unit No. 2

Question 1:

Your proposal states that for the case of a full core discharge, the fuel is moved into the pool four hundred hours after reactor shutdown. If the heat removal capabilities of the fuel pool are to be based on the above assumptions, then the plant technical specifications will have to be modified so that fuel from the core can be removed only after four hundred hours following shutdown. Otherwise, calculating the fuel pool water temperatures using the assumptions in the Branch Technical Position APCSB 9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling," enclosed, we find that the existing system would not be capable of maintaining the water temperature at your design objectives as stated in the FSAR. If this temperature cannot be met, provide the following:

(Three items of information requested.)

RESPONSE

An application has been submitted on July 23, 1975 requesting that Technical Specification 3.8 be amended to , in part, permit unloading of a full reactor core only after four hundred hours have elapsed following shutdown.

Question 2:

Re-evaluate the spent fuel pool accident and dropping of the fuel cask accident taking into consideration the closer spacing for the proposed spent fuel locations. Specifically, justify the decontamination factors used due to the higher pool temperatures.

RESPONSE

With the restrictions referred to in the response to Item 1, the spent fuel pool water tempeature will be no higher than the design objective stated in the FSAR, therefore, no new decontamination factors would apply to the evaluation of the spent fuel accident. In all other respects, as indicated in Reference 2, in the response to Item A.5 of Reference 1, the spent fuel accident would not be changed with the planned modification, and need not be re-evaluated.

The accidental dropping of a fuel cask onto spent fuel was not evaluated in the FSAR. The required amendment to Technical Specifications, referred to in Item 1, includes a restriction that fuel casks may not be moved over the spent fuel pit, if the pit contains spent fuel, for the first forty-five days following shutdown for refueling. An evaluation performed using the principal assumptions outlined in NRC Regulatory Guide 1.25 shows that even with damage to the maximum number of fuel assemblies that could be damaged by a fuel cask dropped into the spent fuel pool, the exposure limits of 10CFR100 would not be exceeded if forty-five days elapsed after shutdown.

References

1. Letter from G. Lear to W. J. Cahill, dated April 2, 1975
2. Letter from W. J. Cahill to G. Lear, dated May 9, 1975.

Attachment B

Information Requested by Division of Reactor Licensing (by telephone on June 10, 1975)

Indian Point Unit No. 2

Item 1 - Of what material will new racks be constructed?

RESPONSE:

The new racks will be constructed of Type 304 stainless steel A-240 or A-276 as indicated on drawing included in May 9, 1975 submittal.

Item 2 - What design criteria are used since this material is not included in the referenced code (American Institute of Steel Construction)?

RESPONSE:

The ratios of allowable load to yield strength given in the AISC code for carbon steels are applied to the yield strength of the stainless steel to obtain the allowable loads.

Item 3 - What are the minimum yield strength and the modulus of elasticity of this material?

RESPONSE:

For Type 304 stainless steel, the minimum yield strength is 30,000 psi and the modulus of elasticity at $70^{\circ}F$ is 28.3×10^{6} psi.