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October 23, 1989

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

Document Control Desk
U.S. Nuclear Regulatory Commission
Mail Station P1-137
Washington, D.C. 20555

SUBJECT: Indian Point Unit No. 2 Spent Fuel Pool Storage Capacity Increase (TAC 72962)

REFERENCES: 1. Letter to Mr. Stephen B. Bram from Mr. Donald S. Brinkman, dated September 18, 1989 entitled: "Request For Additional Information On Spent Fuel Pool Rerack"

Reference 1 requested Consolidated Edison to provide additional information regarding storing Indian Point Unit No. 2 spent fuel. Attachment I to this letter contains the radiological information requested in Reference 1.

Should you or your staff have any questions regarding this matter, please contact Mr. Jude DelPercio, Manager, Nuclear Safety and Licensing.

Very truly yours,

Stephen B. Bram

Add: NRR/DREP/RPB 4+1 Encl

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ATTACHMENT I

ADDITIONAL INFORMATION REGARDING RADIOLOGICAL PROTECTION FOR THE INDIAN POINT 2 INCREASE IN SPENT FUEL POOL STORAGE CAPACITY

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
OCTOBER, 1989

QUESTION #1

Provide calculations showing the maximum radiation dose rates in accessible areas adjacent to the Spent Fuel Pool (SFP) resulting from this change. Calculation should be based on conservative assumptions (i.e., a 'fresh' core offload stored in a rack location that has the smallest distance to the pool wall). Do the results conform to the radiation zoning shielding design criteria for the plant?

RESPONSE

The proposed increase in the spent fuel storage capacity of the Indian Point 2 Spent Fuel Pool will not affect dose rates external to the pool wall. This is due to the fact that the majority of the Spent Fuel Pool is below ground level, therefore there are no accessible areas adjacent to the fuel storage racks (see FSAR section 1.2). The closest accessible areas adjacent to the pool walls are a minimum of approximately 10.3 feet above the stored fuel. In addition, the concrete pool walls adjacent to these accessible areas are approximately 6.25 feet thick. Ignoring the water, the concrete provides approximately 12 feet of shielding on the most direct line from the nearest point of accessibility to the top of the stored fuel for the proposed rerack. In comparison, there is approximately 11 feet of concrete shielding for the most direct line with the existing storage racks. Therefore, the shielding provided by the water and concrete is such that no significant change in dose rates in accessible areas is expected for the proposed rerack.

With regard to dose rates at the edge or above the spent fuel pool, no change is expected. Operating experience normally shows dose rates of less than 2.5 mRem/hr in these areas regardless of the quantity of fuel stored. This is not expected to change with the proposed reracking because radiation levels in these areas are due primarily to radioactivity in the water, which our experience has shown returns to a level of equilibrium following fuel movement. Stored fuel is so well shielded by the water above the fuel that dose rates at the top of the pool from this source are negligible.

Based on the above, there are no radiation zones related to the shielding design criteria for the spent fuel pool that would be affected by the proposed reracking.

QUESTION #2

Industry experience has shown that it is very difficult to rerack a SFP without the use of divers. In light of this, justify your assumption that no divers will be needed at Indian Point 2. If divers will be used, provide an estimate of the radiation exposure to the divers expected and discuss the radiation control measures (such as those discussed in NRC Information Notice 82-31) that will be employed to ensure these exposures are ALARA.

RESPONSE

The existing spent fuel storage racks consist of twelve independent free standing modules. In addition, the proposed new spent fuel storage racks consist of twelve independent free standing modules. Consequently, the use of divers to uncouple rack modules or attach handling rigging is not necessary. In the unlikely event that problems should arise during reracking operations requiring the use of divers, the following considerations would govern operations to assure that exposures are maintained ALARA:

1) Spent fuel will be relocated to minimize radiation exposure to the divers.

2) Pre-dive planning meetings will be held with Health Physics personnel and divers to insure that exposure received while working is ALARA.

3) Procedures associated with diving will be reviewed to insure incorporation of ALARA techniques and radiological controls.

4) Whenever fuel movement occurs, QA personnel will independently witness and verify the new locations of the fuel assemblies. Other irradiated objects greater than 1 R/hr on contact will be controlled in a similar manner. After any movement of either fuel or irradiated components (>1 R/hr), an underwater radiation survey will be conducted before any diving operations resume.

5) Daily, before any diving operation, a radiation survey of the diving area will be conducted. This survey will be performed using two independent radiation exposure monitoring devices. A survey map of the pool will be updated to reflect current status of ongoing fuel rack modification.

6) Each diver will be equipped with a calibrated, alarming dosimeter; this dosimeter will be checked each day before diving operations begin. Each diver will also be equipped with a remote-readout radiation detector which will be continuously monitored by health physics technicians. The divers will surface and have their dosimetry checked periodically; any significant deviation from the expected dive work pattern or radiation levels will be grounds for dive termination.

7) Fuel pool clarity and underwater lighting acceptance criteria will be established to help ensure adequate visibility is maintained.

8) Hot particle controls will be in effect as required.

The above radiation control measures are in accordance with NRC Information Notice 82-31.

The use of divers, if needed, during the rerack is expected to be very limited. There are very specific station procedures developed for diving and for hot particle control. A conservative estimate for diver radiation exposure should divers be necessary is less than 1 person rem based on a total of 40 person hours of work in the spent fuel pool.

QUESTION #3

Perform an analysis of the radiological consequences of postulated fuel handling accident (NUREG-0800, "Standard Review Plan," Section 15.7.4) and spent fuel cask drop accidents (NUREG-0800, "Standard Review Plan," Section 15.7.5) inside the containment.

RESPONSE

At Indian Point 2, the Fuel Storage Building and the Containment Building are separate structures. Consequently, fuel casks would only enter and be handled in the Fuel Storage Building. Consideration of a spent fuel cask drop accident in containment, therefore, is unnecessary. Additionally, the Indian Point 2 Technical Specifications prohibit movement of a spent fuel cask over the Spent Fuel Pool at any time. Therefore, in accordance with Technical Specifications, the drop of a spent fuel cask in the Spent Fuel Pool is precluded.

An analysis of the consequences of a postulated fuel handling accident in the Fuel Storage Building is presented in Section 8 of Attachment B to Con Edison's June 20, 1989 request for a license amendment to expand spent fuel storage capacity. This analysis is in conformance with NUREG-0800, Section 15.7.4 and Regulatory Guide 1.25. Since, as stated above, a cask drop in the Spent Fuel Pool is precluded, this analysis was not included in the June 20, 1989 submittal. For purposes of completeness, a cask drop analysis in the the Spent Fuel Pool was done as part of the previous rerack assessment. That cask drop analysis very conservatively assumed the complete failure of all fuel rods in an entire core loading (193 assemblies) at 90 days after shutdown with a power level of 3216 Mwt. In response to your request, we have again performed the cask drop analysis using the same basic methodology with updated core inventories as determined by the ORIGEN II code (see the June 20, 1989 submittal for radionuclide inventories). The results of this reanalysis are:

Inhalation Thyroid Dose	2.1 Rem
Beta Skin Dose	18.2 Rem
Whole Body Gamma Dose	0.16 Rem

Con Edison has separately requested a license amendment to increase the authorized power level for Indian Point 2. There were several submittals regarding this amendment request. The March 30, 1989 submittal [letter from Mr. Stephen B. Bram to USNRC Document Control Desk entitled: "Supplemental Application for License Amendment to Increase Authorized Power Level (TAC No. 69542)] provides the containment fuel handling accident information. As part of that request, results of an analysis of a postulated fuel handling accident in containment were submitted. The results of this analysis are:

Thyroid Dose	100 Rem
Whole Body Gamma Dose	0.4 Rem

The assumptions used to determine these values are presented in Table 3-1. Since the proposed rerack does not impact fuel handling operations inside containment, this analysis is not affected and remains valid.

Table 3-1
Assumptions For The Evaluation
Of The Fuel Handling Accident In Containment

1.	Power level, Mwt	3216
2.	Radial Peaking Factor	1.65
3.	Gap Fractions	per Reg. Guide 1.25
4.	Number of Failed Assemblies	1
5.	Filters Assumed	None
6.	Pool Decontamination Factor for Iodine	100
7.	Decay Time Before Moving Fuel, hr	174
8.	Atmospheric Dispersion (X/Q)	5.2×10^{-4} sec/m^3 (per Tech Spec 3.8.B.4 bases)

QUESTION #4

If changed from previous assessment provide the number of spent fuel assemblies that could be damaged by dropping each typical load carried over the pool.

RESPONSE

There are no significant changes in the number of spent fuel assemblies that could be damaged from those postulated in the previous rerack assessment.

The minimum distance between the stored fuel assemblies and racks being handled has been increased from less than three feet for the previous rerack to four to six feet (depending upon the pool arrangement during a rack move) for the current proposed rerack. Therefore, fewer fuel assemblies have a potential to be damaged from a rack drop than in the previously approved rerack assessment.

The Indian Point 2 Technical Specifications prohibit movement of a rack directly over fuel assemblies. In the unlikely event that a rack was dropped and damaged fuel assemblies, the cask drop analysis presented in response to Question 3 above is bounding. The only other typical load carried over the Spent Fuel Pool is a fuel assembly which is discussed in our response to Question 3, above.