Docket No. 50-286

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JUL 22 1975

Mr. William J. Cahill, Jr.
Vice President
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

Dear Mr. Cahill:

This letter is in regard to a recently identified problem associated with the calculation of loads on the reactor pressure vessel supports that may result as a consequence of a postulated instantaneous double ended break in the reactor coolant system cold leg pipe at the pressure vessel nozzle. In order to permit us to assess the adequacy of the Indian Point 3 reactor vessel support system for this postulated condition, we request that you provide the information listed in Enclosure 1.

We discussed this problem with your representatives at a meeting held on June 25, 1975 in Bethesda, Maryland. At that meeting, we provided your representatives with a tentative list of additional information that may be required. Our finalized list of required additional information is provided in Enclosure 1.

Please inform us within seven days after receipt of this letter of your schedule for providing this information. Contact us if you have any questions regarding the information requested.

Sincerely,
Original Signed by
Domenic Vassallo

D. B. Vassallo, Chief
Light Water Reactors
Project Branch 1-1
Division of Reactor Licensing

appl. 3

Enclosure:
Request for Additional
Information

cc: See next page

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DATE	7/21/75	7/22/75

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ENCLOSURE 1

INDIAN POINT UNIT 3

DOCKET NO. 50-286

REQUEST FOR ADDITIONAL INFORMATION

Recent analyses have shown that reactor pressure vessel supports may be subjected to previously underestimated lateral loads under the conditions that would exist if an instantaneous double ended break is postulated in the reactor vessel cold leg pipe at the vessel nozzle. It is therefore necessary to reassess the capability of the reactor coolant system supports to limit the calculated motion of the reactor vessel during a postulated cold leg break within bounds necessary to assure a high probability that the reactor could be brought safely to a cold shutdown condition.

The following information is required for purposes of making the necessary reassessment of the reactor vessel supports at Indian Point 3:

1. Provide engineering drawings of the reactor support system sufficient to show the geometry of all principal elements and materials of construction.

2. Specify the detail design loads used in the original design analyses of the reactor supports giving magnitude, direction of application and the basis for each load. Also provide the calculated maximum stress in each principal element of the support system and the corresponding allowable stresses.

3. Provide the information requested in 2. above for the RV supports considering a postulated break at the cold leg nozzle. Include a summary of the analytical methods employed and specifically state the effects of short term pressure differentials across the core barrel in combination with all external loadings calculated to result from the required postulate. This analysis should consider:

   (a) limited displacement break areas where applicable
   (b) consideration of fluid structure interaction
   (c) use of actual time dependent forcing function
   (d) reactor support stiffness.

4. If the results of the analyses required by 3. above indicate loads leading to inelastic action in the reactor supports or displacements exceeding previous design limits provide an evaluation of the following:

   (a) Yield behavior (effects of possible strain energy buildup) of the material used in the reactor support design and the effect on the

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loads transmitted to the reactor coolant system and the back-up structures to which the reactor coolant system supports are attached.

(b) The adequacy of the reactor coolant system piping, control rod drives, steam generator and pump supports, structures surrounding the reactor coolant system, reactor internals and ECCS piping to assure that the reactor can be safely brought to cold shutdown.

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