BEFORE THE UNITED STATES
ATOMIC ENERGY COMMISSION

In the Matter of )

Consolidated Edison Company ) Docket No. 50-247
of New York, Inc. )
(Indian Point Station, Unit No. 2) )

Answers of Applicant to Questions Raised

by Atomic Safety and Licensing Board

on October 5, 1971

Part I



October 28, 1971

# KEY TO QUESTIONS

(B) - Mr. Briggs' Questions

(J) - Chairman Jensch's Questions

The questions have been numbered as follows:

1. Mr. Briggs' question regarding effect of transient on reactor vessel integrity (Tr. 1599-1601)

2. Mr. Briggs' question regarding Staff's check of Westinghouse's ECCS calculations (Tr. 1612)

3. Mr. Briggs' question regarding independent temperature calculation for most severe loss-of-coolant accident (Tr. 1613)

4. Mr. Briggs' question regarding calculation of temperatures by the RELAP Code (Tr. 1613)

5. Mr. Briggs' question regarding assumption of no steam flow through unbroken lines (Tr. 1615-1616)

6. Mr. Jensch's question regarding dissolved oxygen content in the river (Tr. 1616-1618)

Question No. 1 (B) (Tr. 1599-1601)

"Well, we have indicated then that you gentlemen can conclude that the flaws in the reactor vessel, if there are any, any of any significance, it would be highly unlikely that they would be more than half an inch deep by several inches long, and you also conclude, I believe, that a flaw this size would not be sufficient to cause failure of the reactor vessel under the conditions that are imposed by the technical specifications.

"There was a calculation requested, I believe, by the ACRS concerning the situation if there were a transient and the rods failed to drop, in which case, if I recall, the pressures in the reactor vessel would go as high as about 4000 psi. Is this number right? Do you recall the calculation?

"Would you, before the next session that we have, look at that situation and indicate whether the conditions imposed on the reactor vessel by that transient would change any of the statements that you have made in the report here? I don't ask you to do it now. Unless someone is completely familiar with those numbers, that is.

"This additional testimony doesn't include an examination of that transient. There was consideration given when the transient was examined in your other reports of what these vessels would be and whether the reactor vessel would fail. I'd like for you to look at the case of that transient again, the conclusions reached then and what is stated in this report, and see if there is anything different that you put in this report as a result of the conditions that exist in that transient.

"In other words, it is a case that wasn't considered here. I would like to have it looked at again."

Answer:

We have reviewed the analysis and conclusions reported in FSAR Response to Question 14.7 and in Westinghouse Proprietary Report WCAP 74860L to which we assume Mr. Briggs referred. We

Q. 1 (B) (Tr. 1599-1601)

find nothing which changes any previous testimony regarding the integrity of the reactor vessel at Indian Point Nuclear Generating Station Unit No. 2.

Question No. 5 (B) (Tr. 1615-1616)

"[W]hat would the situation be if one changed his assumption so if steam flowed through all four lines and the accumulator water that was entrained in the thing be carried out through the broken loop? In other words, one can make two kinds of assumptions: He can make the assumption that is made in the calculations that there is no steam flow through the three unbroken lines, that the accumulator water that comes into those lines goes into the space around the core barrel and is available at the core; he can make the other assumption that there is steam flow through those lines and that some of the accumulator water is entrained in the steam and is carried out the break. I'd like some discussion about which of these assumptions is the more conservative, and whether there is reason, really good reason for the assumption that is used here rather than the assumption of entrainment of accumulator water in the steam and this water being carried through."

Answer:

The assumption of no steam flow through the unbroken lines during accumulator injection is conservative since steam flow is prevented through the 3 unbroken lines thus greatly reducing the fuel rod heat transfer.

If steam flow is considered to exist through the unbroken lines, the interaction between steam and water in the cold leg pipe will condense the steam thus reducing the pressure and the steam flow velocity. This effect will result in an increased core reflooding rate. Should the liquid be partially entrained by the steam flow, it will separate from the steam in the down-comer region due to the flow area increase, the change in flow direction, and the pressure of the hot leg nozzles.

Q. 5 (B) (Tr. 1615-1616)

It follows that the assumption of no steam flow through the unbroken lines and the associated low values of heat transfer in the core is more conservative, and is the assumption made in the analysis.