UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
CONSOLIDATED EDISON COMPANY OF	)	Docket No. 50-247
NEW YORK, INC. (Indian Point, Unit	)	
No. 2)	)	

REPORT OF CONSOLIDATED EDISON'S INVESTIGATION AND RESOLUTION OF THOSE ISSUES IDENTIFIED AS POTENTIAL UNREVIEWED SAFETY QUESTIONS IN THE LETTER OF THE OFFICE OF INSPECTION AND ENFORCEMENT DATED DECEMBER 11, 1980

January 5, 1981

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INTRODUCTION:

This report is submitted by Consolidated Edison in response to the letter of Mr. Victor Stello, Jr., Directo of the Office of Inspection and Enforcement, dated December 11, 1980. This enclosure presents the results of Consolidated Edison's investigation of the four issues identified in that letter as potential unreviewed safety questions. As requested in Mr. Stello's letter, these matters were considered by Con Edison in light of: (1) plant conditions discovered on October 17, 1980, and (2) plant conditions which could have developed, had the plant again been returned to power without discovery of the leakage and the flooding problems.

On October 17, 1980, an amount of water now believed to have comprised approximately 125 000 gallons was inside the containment of Indian Point Unit 2.

There are a total of 11 level float switches on four separate stalks in two separ e sumps above the top of the curb surrounding the reactor cavity (see page 13-12 December 22, 1980 Consolidated Edison letter to Boyce H. Grier). Each level float switch when actuated would provide a separate and independent indication to the operators in the Control Room. Had the plant been returned to power on October 17, 1980 without discovery of the flooding condition and the leakage, the next level float switch (located in the Recirculation Sump at Elev. 46'-7 1/8")

would have been actuated. Based upon actuation of this level float switch and its associated indication in the Control Room, it is inconceivable that the flooding level would have risen much higher than Elev. 46'-8" without investigation and subsequent reactor shutdown. The amount of water necessary to reach this level inside containment is approximately 150,000 gallons. This condition would result in approximately the bottom 13 feet of the reactor vessel being wetted.

POTENTIAL UNREVIEWED SAFETY QUESTION (1)

"Partial submergence of the hot reactor vessel in cold brackish river water"

Plant Condition 1

See Consolidated Edison's December 22, 1980 letter to Boyce H. Grier, Attachment A, Item 1, which is incorporated by reference, for a description of the plant conditions that were observed in this instance and the effect of immersion on the reactor vessel.

Plant Condition 2

If the plant had been returned to power, and had flooding continued to a level of 13 feet above the bottom of the reactor vessel, no damage would have resulted to the vessel, including the approximately 2-1/2 feet of the lower course of the vessel shell subject to relatively higher neutron irradiation from the lower part of the core.

We have concluded as a result of our examination that such postulated immersion would have no effect upon the integrity of the reactor vessel, and that accordingly this does not constitute an unreviewed safety question.

Westinghouse performed a fracture mechanics evaluation as part of their overall stress analysis for this reactor vessel wetting

incident. The evaluation addressed specifically the lower shell and the bottom head (refer to Section 3.0 of WCAP-9834).

In 1973 Westinghouse evaluated the integrity of the reactor vessel when subjected to external thermal shock, (refer to Appendix 3-A of WCAP-9834) covering the highly irradiated region of a reactor vessel, such as the belt line region. Both of these analyses led to the conclusion that the reactor vessel would be safe from fracture.

For the lower shell and bottom head analysis, Westinghouse used the shell/head region for maximum stress and used the highest of the lower head Reference Temperatures nil ductility transition temperatures (^{RT}NDT's) for conservatism. (Note that the higher the ^{RT}NDT the greater will be the effect on the vessel due to consideration of non-ductile failure). The neutron fluence level for the lower shell and the bottom head are very low and therefore of no concern with respect to brittle fracture; ^{RT}NDT is therefore not affected. Based on an assumed realistic vessel outer surface temperature of 200 F, the postulated critical flaw sizes (on the outside of the reactor vessel) were evalulated to be greater than 30% of the corresponding wall thicknesses and, flaws of this size would certainly be detected during pre-service or inservice examinations of the vessel. The result of the analysis indicates the vessel lower shell and bottom head would be safe from fracture.

To further substantiate the conclusion of the analysis, the Westinghouse 1973 analysis showed that the integrity of the reactor vessel in the highly irradiated region, such as the belt line region, would not be impaired, since the hypothetical critical flaw size was estimated to be a significant fraction of the wall thickness. For the purpose of comparison, the Westinghouse 1973 report used ^{RT}NDT = 60 F and a neutron fluence level of 3.5×10^{19} neutrons/cm^2, which are more severe than the corresponding values for Indian Point 2, which are 34 F and 1.8×10^{19} neutrons/cm^2 respectively, at the end of vessel life, after 32 full power years of service.

POTENTIAL UNREVIEWED SAFETY QUESTION (2)

"Partial submergence of the stainless steel incore instrument conduits in brackish river water"

Plant Condition 1

See Consolidated Edison's December 22, 1980 Letter to Boyce H. Grier, Attachment A, Item 1, which is incorporated by reference, for a description of plant conditions that were observed in this instance and the effect of partial submergence of the stainless steel incore instrument conduits in brackish river water.

Plant Conditon 2

If the plant had been returned to power without discovery of the leakage and flooding problems, the stainless steel incore instrument conduits would have continued to be partially submerged in the brackish river water. The chloride content of the water was approximately 3000 ppm, and the temperature of the water was approximately 100 F. It is conceivable that if flooding continued, the water temperature could have increased.

The Company has concluded as a result of its examination that such postulated continued exposure would have had no effect on the stainless steel instrument conduits, and that, accordingly, this does not constitute an unreviewed safety question.

References indicate that austenitic stainless steels, such as the instrument conduits, do not crack even in strong chloride environments at ambient temperatures or at elevated temperatures when stresses are low. (1) (2) (3).

Laboratory tests have demonstrated that even after 840 hours exposure to boiling 30,000 ppm sodium chloride solution, there were no cracks in austenitic stainless steel. (4). It was also demonstrated that in boiling magnesium chloride (420,000 ppm chloride, 310 F), an applied stress of 25,000 psi was required to cause cracking in 18/8 stainless steel in 60 hours. (5). The temperature of the sump water could not exceed 212 F, and the stress applied to the instrument conduits was less than 5000 psi.

Dye penetrant tests of the conduits after the flooding indicated that there were no defects. Continued exposure to the flooding environment is not considered aggressive enough to result in cracking of austenitic stainless steel regardless of the length of time of immersion. Consequently, it is concluded that no damage to the instrument lines would have resulted from the hypothetically assumed continued exposure.

References:

(1) Logan, M. L. "The Stress Corrosion of Metals" John Wiley and Sons, NY (1966)

(2) Miller, G. E. "Designing with Stainless Steel for Service

in Stress Corrosion Environments" Materials Performance (May 1977)

(3) Robertson, W. D., "Stress Corrosion Cracking and Embrittlement" John Wiley and Sons, NY (1956)

(4) Edeleanu, C., Journal of the Iron and Steel Institute, 173, (1953)

(5) Hines, J. G. and Hoar, T. P., Journal of the Iron and Steel Institute 184 (1956)

POTENTIAL UNREVIEWED SAFETY QUESTION (3)

"Potential post-Loss of Coolant Accident (LOCA) water levels in containment in excess of the assumptions used in the Safety Analysis Report (SAR)"

In 1976, Con Edison performed an evaluation of flooding inside the Unit 2 containment building after a postulated design basis loss of coolant accident (LOCA). This evaluation was reviewed by NRC and approved on September 4, 1976 in Amendment No. 20 to the Indian Point 2 Facility Operating License No. DPR-26. The amount of water inside containment as a result of the actuation of the emergency core cooling system following a LOCA was determined to be approximately 423,000 gallons. This amount of water would have reached approximately Elev. 50'-1" inside containment. The water level would have to reach Elev. 50'-5" before any electrical safeguard components required for post-LOCA operation would be submerged. The data generated by our 1976 study has been employed in conjunction with the two plant conditions set forth in the December 11, 1980 Stello letter in order to reach the following conclusions.

We have concluded as a result of our examination that neither of the postulated conditions would result in any safeguard functions being rendered inoperable, and that accordingly this does not constitute an unreviewed safety question.

Plant Condition 1

In the very unlikely event that a LOCA (large break) had occurred in conjunction with a plant condition like that found on October 17, 1980 (approximately 125,000 gallons of water already on the floor), a total of approximately 548,000 gallons would accumulate inside containment. The resultant water level would reach Elev. 51'-7 1/2". This would result in the submergence of (1) safety injection valves 856A, B, C, E and F, and (2) the second tier of electrical penetrations.

Cold leg injection valves (856A, C and E) are required to be open to provide cooling water to the core during a LOCA. These valves are normally open, receive a confirmatory safety injection signal to open, and are designed to "fail as is". Therefore, submergence of these valves would not impede operation of the core cooling system.

Hot leg safety injection valves (856B&F) are de-energized in the closed position. At approximately 24 hours after the LOCA, a hot leg injection path would be needed to assure boron precipitation does not occur. Even if these valves were submerged and could not be opened, there are two other hot leg injection paths available. These paths are shown in a February 19, 1976 Con Edison letter to the NRC (William J. Cahill, Jr. to Robert W. Reid) on submerged valves.

The bottom of the Recirculation Pump motors are at the 52'-5" elevation and would not be affected by this hypothetical flooding/LOCA condition.

The electrical splices located at the second tier of electrical penetrations would also be wetted. Since these splices and penetrations were designed and tested for accident conditions which included wetting under 100% R.H. conditions at 271 F and 47 psig, they would not be adversely affected.

Plant Condition 2

If a large break LOCA has occurred in conjunction with a plant condition like that which might have developed because the flood condition and the leakage had not been discovered and the plant returned to power (approximately 150,000 gallons of water on the floor), a total of approximately 573,000 gallons would accumulate inside containment. The resultant water level would reach Elev. 51"-11". No additional electrical safeguard components required for post-LOCA operation, beyond those already discussed in the evaluation of plant condition 1, would be affected.

POTENTIAL UNREVIEWED SAFETY QUESTION (4)

"Potential Post-LOCA water boron concentrations less than the assumptions used in the SAR"

Plant Conditions 1 and 2

An evaluation has been made of post-LOCA water boron concentrations, hypothetically assuming a LOCA had occurred in conjunction with (1) plant flood conditions discovered on October 17, 1980, and (2) plant conditions which could have developed, had the plant again been returned to power without discovery of this leakage and flooding problems.

This evaluation was based on minimum required water inventory in the boron injection tank and the refueling water storage tank and on maximum water inventory in the spray additive tank, thus resulting in a conservative estimate of boron concentration. With these borated water sources available during a LOCA, the amount of unborated water required for the system to go critical is approximately 950,000 gallons. This is far in excess of the approximately 125,000 gallons of unborated water presented in assumed plant condition 1 and the approximately 150,000 gallons of water assumed in plant condition 2. Therefore, it is concluded that a return to criticality would not occur following a LOCA in conjunction with containment flooding as hypothetically assumed in either plant condition 1 or 2, and that accordingly this does not constitute an unreviewed safety question.