Assistant Vice President

Consolidated Edison Company of New York, Inc.
4 Irving Place, New York, N Y 10003
Telephone (212) 460-2533

January 5, 1981

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

Mr. Victor Stello, Jr., Director
Office of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Consolidated Edison's Response to Notice of Violation and Proposed Imposition of Civil Penalty dated December 11, 1980

Dear Mr. Stello:

This is in response to your letter of December 11, 1980, enclosing a Notice of Violation and Proposed Imposition of Civil Penalty resulting from your office's investigation of the recent accumulation of service water inside containment at Indian Point Unit 2. Your letter also proposed civil penalties characterized as based upon the October 7, 1980 Interim Enforcement Policy and remarked generally upon the management control system at Indian Point Unit 2. Pursuant to Consolidated Edison Corporate Instruction 250-1, your letter has been referred to me for reply.

It is apparent that on a number of matters we have drawn differing conclusions based upon information gathered during our independent investigations of the October event. Consolidated Edison has conducted its own extensive inquiry into the incident, and our comprehensive report on the event is in the final stages of preparation. While much has been and will continue to be done by us to prevent recurrence of an incident of this sort, we do not agree with the Office of Inspection and Enforcement's suggestion that the management control system at Indian Point Unit 2 was not functioning in an appropriate manner. The confluence of equipment failures necessary to permit the occurrence of this incident could not reasonably have been anticipated from events prior to October 17, 1980, and there were, in our judgment, no management system failures -- much less violations.

A report on Consolidated Edison's management control system at Indian Point Unit 2 was issued by your Office of Inspection and Enforcement on September 2, 1980, following an in-depth audit of our central and plant operations in July and August.

Its conclusions on our attitude towards safety at Indian Point note that:

> "It was the general view that management interpreted the technical specifications literally or conservatively (with respect to safety) and that plant operators had no reservations about shutting the plant down if, in their opinion, technical specification limits or other safety considerations required it. Based on the above interviews, indications were that plant and corporate management's first priority was safe operation and that the operating staff was aware of this priority."

Our differing views on these and several other matters relating to the event were arrived at only after a careful consideration of the information and conclusions set forth in your December 11 letter and its enclosed Notice. Our consideration has been substantially aided by the many studies and analyses which we and others have performed since October. Our responses are set forth in two enclosures to this letter: Consolidated Edison's Statement in Reply to Notice of Violation, and Consolidated Edison's Answer to Notice of Proposed Imposition of Civil Penalty. We also enclose a third document which addresses the four alleged unreviewed safety questions referred to in your December 11 letter.

Our major differences of view can be summarized as follows: (1) our use of epoxy materials in repairing service water piping was prudent and effective, and not a contributor to the event; (2) the maintenance and failure detection program for service water system leakage, in conjunction with the design of related plant systems, was appropriate for the infrequent pinhole leaks which had been experienced at the unit during its prior operating history; (3) our conclusion that there were no problems with the safe operation of the plant at any time during the event is correct and has in fact been borne out by our analyses, even though your findings of violation, and in particular your proposed severity levels, implicitly assume that there was a high degree of potential impact on the public; and (4) while we have previously acknowledged that the events discovered on October 17 should have been promptly reported to the NRC, the NRC reporting requirements as set forth at the time of the event were unclear, and in light of your present claims, provided an inadequate level of guidance for operator conduct.

The Notice of Violation does not acknowledge that certain operational conditions or circumstances now perceived by your Office as problems were simply unrecognized prior to the event, and thus not addressed either by NRC regulations or by key unit operational documents. Many of the regulatory requirements now cited as the basis for violations are in our judgment ill-suited to support the Office of Inspection and Enforcement's violation

claims. As such, we believe that your Office has neglected a key provision of the Interim Enforcement Policy:

> "Corrective enforcement actions may be taken in the absence of any violation of NRC requirements; for example, when a safety problem not previously covered by a requirement is discovered. NRC imposes civil penalties, however, only on the basis of a violation of an existing requirement. "(45 FR at 66755).

Our careful review of the Notice of Violation leads us to the conclusion that there were other instances where the new (October 7, 1980) Interim Enforcement Policy was not followed. In particular, in both concluding that violations existed, and in assessing severity levels, the Notice of Violation accords vague and, in this instance, unsupportable interpretations to such broad terms as "controlled or expected condition," "abnormal degradation in containment," "modification," and "actual or high potential impact on the public," none of which could reasonably have been so interpreted by nuclear power plant operators prior to the occurrence.

The Notice also misapplies severity levels by assuming actual safety implications existed as a consequence of the event when in fact our investigations revealed none. Lastly, the Notice unfairly and improperly applies footnote 17 of the Interim Enforcement Policy (45 FR at 66757) to "bootstrap" lesser alleged violations into higher categories and to create multiple violations out of single events without a proper basis. Because of the importance to industry enforcement generally of the Office of Inspection and Enforcement's unprecedented (and in our view unwarranted) interpretation of the Policy in this instance, we have taken the liberty of sending a copy of this letter to the Commission Secretary and the Executive Director for Operations, in order that it may be considered in connection with the current round of solicitation of comment on the Interim Enforcement Policy.

Although we are unable to concur with many of the conclusions of the Office of Inspection and Enforcement that violations occurred, it is clear to us that a revised and heightened consideration of various potential implications of service water system leakage should be made by Consolidated Edison and the nuclear industry generally. To identify the nature of the desired changes, we have undertaken an extensive internal review of management operations in the area of component failure assessment. While many of the numerous changes intended to prevent recurrence have been previously provided to the NRC for review, and most recently in our letter of December 22, 1980, it may be useful to mention them again here. Programmatic changes are planned, such as revisions to the Quality Assurance Program and redefinition of certain of the functions of the

Station Nuclear Safety Committee. Implementation changes include revisions to notification, maintenance and surveilance procedures. These changes and further training of personnel to accomplish them will result in additional improvements in the management control system for Indian Point Unit 2.

In addition, we are reviewing our existing organizational structure for operation and operations support at Indian Point in light of recent guidance (September 23, 1980; Vargas to Zarakas) issued by the NRC, to determine if changes could beneficially be made. To date, one such change has been identified as warranted. In several instances, the management control concern centers on implementation of the 10 CFR 50 Appendix B quality assurance program. This program at Con Edison could be beneficially modified by a re-organization of the organizations responsible for implementation and oversight. We plan to have all the Indian Point QA/QC function be part of the Central Quality Assurance and Reliability Department which reports to offsite management independent of power generation.

These changes and our commitment to continuous improvement in the operation of Indian Point Unit 2 demonstrate our intent to implement whatever actions appear appropriate to prevent recurrence of the matters outlined in your letter. We want to assure you of our firm resolve to take all necessary steps to continue our eighteen-year record of safe operation of nuclear power plants at Indian Point. We look forward to working with the appropriate offices of the Commission in this effort.

Very truly yours,

John D. O'Toole
Assistant Vice President

Enclosures

cc: Mr. Samuel J. Chilk, Secretary to the Commission
Mr. William J. Dirks, Executive Director for Operations
Mr. Boyce H. Grier, Director Region I
Mr. Theodore Rebelowski, Resident Inspector

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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

CONSOLIDATED EDISON'S
ANSWER TO NOTICE OF PROPOSED
IMPOSITION OF CIVIL PENALTY

Pursuant to 10 CFR 2.205(b) and the Notice of Violation and Proposed Imposition of Civil Penalty dated December 11, 1980, Consolidated Edison Company of New York, Inc., provides the following written answer. The answer to each alleged violation incorporates by reference the corresponding response to that item set forth in Consolidated Edison's Statement in Reply to the Notice of Violation (hereinafter referred to as "Statement").

Alleged Violation I.A: Consolidated Edison admits that this item is a violation. Consolidated Edison denies that this item is a Severity Level III violation under the October 7, 1980 Interim Enforcement Policy (hereinafter referred to as "Policy"). Consolidated Edison believes that there are extenuating circumstances and requests remission or mitigation of any penalty proposed in accordance with the Policy. See Statement, pp. 2-11.

Alleged Violation I.B: Consolidated Edison denies that this item is a violation. See Statement, pp. 13-16.

Alleged Violation II.A: Consolidated Edison admits that this item is a violation. Consolidated Edison denies that this item is a Severity Level III violation under the Policy. Consolidated Edison believes there are extenuating circumstances and requests remission or mitigation of any penalty proposed in accordance with the Policy. See Statement, pp. 18-22.

Alleged Violation II.B: Consolidated Edison denies that this item is a violation. See Statement, pp. 26-37.

Alleged Violation II.C: Consolidated Edison denies that this item is a violation. See Statement, pp. 39-42.

Alleged Violation II.D: Consolidated Edison denies that this item is a violation. See Statement, pp. 45-50.

Alleged Violation II.E: Consolidated Edison denies that this item is a violation. See Statement, pp. 53-55.

Alleged Violation II.F: Consolidated Edison admits that this item is a violation. Consolidated Edison denies that this item

is a Severity Level III violation under the Policy. Consolidated Edison believes that there are extenuating circumstances and requests remission or mitigation of any penalty proposed in accordance with the Policy. See Statement, pp. 58-60.

Alleged Violation III.A: Consolidated Edison denies that this item is a violation. See Statement, pp. 63-67.

Alleged Violation III.B: Consolidated Edison admits that this item is a violation. Consolidated Edison denies that this item is a Severity Level III violation under the Policy. Consolidated Edison believes that there are extenuating circumstances and requests remission or mitigation of any penalty proposed in accordance with the Policy. See Statement, pp. 68-70.

Alleged Violation IV: Consolidated Edison admits that this item is a violation. Consolidated Edison denies that this item is a Severity Level V violation under the Policy. See Statement, pp. 72-73.

Affirmative Defense No. 1: The December 11, 1980 Notice of Violation and Proposed Imposition of Civil Penalty does not apply the Policy according to its terms.

Affirmative Defense No. 2: The NRC was aware of and did not object to Consolidated Edison's program of maintenance and detection of leaks in its service water system at Indian Point 2 as it existed prior to October, 1980 by review and regulatory oversight of the program and associated maintenance history documentation.

Affirmative Defense No. 3: The NRC's Policy as applied to Consolidated Edison herein is vague and indefinite, and does not give Consolidated Edison adequate notice of the standards by which its conduct is to be judged.

Affirmative Defense No. 4: The NRC's Policy as applied to Consolidated Edison herein is punitive, and is not confined to remedial purposes.

Based upon the foregoing answer, Consolidated Edison respectfully requests that the Office of Inspection and Enforcement dismiss those alleged violations which are denied herein, and reduce the cumulative amount of the remaining civil penalties which have been proposed.

Respectfully yours,

John D. O'Toole

John D. O'Toole
Assistant Vice President
Consolidated Edison Company
of New York, Inc.

Dated: New York, New York
January 5, 1981

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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

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

INTRODUCTION:

This report is submitted by Consolidated Edison in response to the letter of Mr. Victor Stello, Jr., Director 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 separate 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.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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

CONSOLIDATED EDISON'S
STATEMENT IN REPLY TO
NOTICE OF VIOLATION

In accordance with 10 CFR 2.201, and the Office of Inspection and Enforcement's Notice of Violation and Proposed Imposition of Civil Penalty dated December 11, 1980, Consolidated Edison Company of New York, Inc., licensee of Indian Point Unit 2, supplies the following responses to the alleged items of noncompliance with NRC regulations set forth in the Notice.

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OFFICE OF INSPECTION AND ENFORCEMENT STATEMENT OF ALLEGED VIOLATION

I. The Commission regulations and the facility license require the licensee to report occurrences important to safety as indicated below.

A. 10 CFR 50.72(a), "Notification of significant events", requires that:

"Each licensee of a nuclear power reactor, licensed under para. 50.21 or para. 50.22 shall notify the NRC Operations Center as soon as possible and in all cases within one hour by telephone of the occurrence of any of the following significant events and shall identify that event as being reported pursuant to this section:

(3) Any event that results in the nuclear power plant not being in a controlled or expected condition while operating or shutdown."

Contrary to the above, the following condition was not reported within one hour of identification:

The discovery on October 17, 1980 of unexpected conditions not specifically considered in the safety analysis report or technical specifications that required remedial action to prevent existence or development of an unsafe condition, specifically the existence of: a flooded reactor vessel pit, about four inches of river water on the vapor containment floor, and steam exiting the instrument thimble holes.

The containment flooding condition was found on October 17, 1980, but not reported to the NRC until October 20, 1980, which did not comply with the one hour reporting requirements of 10 CFR 50.72. Each day that the violation continued constitutes a separate violation for the purpose of computing the civil penalty.

This is a Severity Level III violation (Supplement I.C.2 of the Interim Enforcement Policy). Applying the civil penalty for each day that the violation continued results in a civil penalty of - $120,000.

DISCUSSION:

The existence of water at the bottom elevations of the Indian Point 2 containment building does not necessarily result in the nuclear power plant "not being in a controlled or expected condition," as those terms are used in 10 CFR 50.72(a). By design, water is collected on the containment floor from various sources within containment, the fan cooler units being one such source. The containment floor is designed for the very purpose of directing water via the floor trench system to the containment sump, which is equipped with two sump pumps to transfer the water out of containment to the liquid waste processing system.

Because of the physical location of the reactor cavity (i.e., the lowest point in the containment building), it is reasonable to assume that water could accumulate there also. Therefore, the containment floor was built with curbs around the openings to the reactor cavity to prevent water, which could collect on the containment floor, from entering the cavity under most circumstances. Consolidated Edison installed two submersible sump pumps in the reactor cavity after the plant went into service (i.e., they were not part of the original licensed plant design), to facilitate removal of water from the cavity should it collect there during refueling.

During the period October 17-20, 1980, Consolidated Edison operating personnel knew that there was water on the containment floor and in the reactor cavity but did not recognize that the reactor vessel had been wetted. Both the major source of the water (service water) and the reason for the accumulation of water (both containment sump pumps malfunctioning) were known and appropriate action was promptly initiated to rectify the situation. The Indian Point 2 Technical Specifications do not place any specific limitations on fan cooler unit leakage into containment during operation. Such leakage has been promptly identified and corrected when it has occurred. In addition, the level of water discovered in containment at the time of the event was well below that where safety-related electrical components would begin to become submerged.

The Indian Point Unit 2 FSAR also recognizes that there may be occasional leaks in the fan cooler units, since provisions are incorporated in the plant design to detect fan cooler leakage and to isolate leaking units when necessary. During normal operation, leakage from the fan cooler unit cooling coils is detected along with condensate from the containment atmosphere by the fan cooler unit condensate weir measuring system. During accident conditions, redundant radiation monitors would detect high radioactivity in the fan cooler service water discharge line. If fan cooler leakage did result in such an indication on radiation monitors, the appropriate cooler(s) would be isolated.

Because of the unit's design and the safety procedures described above, at no time during the event was the plant considered to be in an uncontrolled or unsafe condition, either while the removal of water from containment was in progress, or during the investigation of the level of water in the reactor cavity.

Only the reactor trips on October 17 were believed to be reportable to NRC. These notifications to NRC were made in a timely manner; however, the underlying cause was not known and, therefore, could not be transmitted at that time.

When, on October 20, 1980, Consolidated Edison's investigation yielded a concern for potential long-term chloride stress corrosion of the incore instrumentation conduits, a decision was made to perform an examination as to the acceptability of these conduits for continued service, and the NRC was notified. Water level estimates made on the 20th, based on water inventories and chloride swipe samples, did not demonstrate that the water level had reached the bottom of the reactor vessel.

From October 20, 1980 on, there was open communication with the resident inspector and members of the NRC staff at Region 1 Office. When, on October 24, 1980, evidence demonstrated that the reactor vessel had been wetted, the NRC was again immediately informed.

In promulgating 10 CFR 50.72, the NRC provided no definition of what an "uncontrolled" or "unexpected" condition is. However, the NRC Regulatory Staff issued IE Information Notice No. 80-06 on February 27, 1980, immediately prior to the effective date of the 10 CFR 50.72 regulation, which provided some information regarding the intent of the new regulation. This document states that certain enumerated "significant" events were to be reported within one hour. Additionally, it states that significant events comprise:

> "...serious events that could result in an impact on public health and safety such as those leading to initiation of the licensee's emergency plan or any section of the plan, the causing of a nuclear power plant to be in an uncontrolled condition, the exceeding of a safety limit, an act of sabotage, or an uncontrolled release of radioactivity".

The Information Notice also discusses the same requirement contained in 10 CFR 50.72(b) that once any event results in the nuclear power plant not being in a controlled or expected condition while operating or shutdown:

> "[T]he licensee, in addition to prompt telephone notification, shall also establish and maintain an open, continuous communication channel with the NRC Operations Center, and shall close this channel only when notified by NRC."

On July 29, 1980, a Supplement to IE Information Notice 80-06 was issued by the Regulatory Staff because "[e]xperience with notifications being made in accordance with [10 CFR 50.72] suggest the need for clarification and more definitive

guidance" in implementing that notification requirement.

The Supplement states that:

> "Commenters particularly stressed the need to establish more definitive thresholds for notifications of certain types of events... . The general categorization of some of the event types listed in the rule has resulted in notification of events of less significance than originally intended... . The rule was intended to require immediate notification of 'serious events that could result in an impact on the public health and safety.'"

The IE Information Notice Supplement thus reinforces the guidance provided by the original Notice that only "serious events" are intended to be reported under 10 CFR 50.72.

The conditions discovered at Indian Point 2 on October 17, 1980 did not represent a "serious event" in the context of the Information Notice 80-06 clarification, and certainly did not warrant an "open, continuous communication channel" with NRC. This is confirmed by the fact that once NRC was notified of the accumulation of water in containment on October 20, 1980, they saw no need for establishing and maintaining an open communication channel. Given the conditions as they knew them to exist, plant personnel reasonably concluded that the plant was in a safe, controlled condition and certainly did not present any imminent threat or prospect of threat to public health and safety during the period October 17-20, 1980.

Although the accumulation of water on the containment floor is to be expected, the actual amount of water discovered in the containment by plant personnel on October 17, 1980 did nonetheless represent, in the language of the regulation, an "unexpected condition", and in retrospect should have been promptly reported to the NRC. On this basis, we do not contest that a violation occurred as stated in paragraph I.A of the Notice of Violation, as the Office of Inspection and Enforcement has here interpreted 10 CFR 50.72.

Consolidated Edison submits, however, that the 10 CFR 50.72 regulation is vague and indefinite with respect to what constitutes "uncontrolled" or "unexpected" plant conditions. Guidance provided to date by NRC still has not established a general consistency in interpretation of the rule, and Consolidated Edison is unaware of any prior instance where the NRC has construed "unexpected condition" in a one-hour reporting context. Accordingly, the regulation provides a questionable basis for the retrospective evaluation of operations reporting, in that the licensee is not accorded sufficient information to reasonably determine the standards by which its conduct will be adjudged.

The reporting standard by which this event is being retrospectively considered is ambiguous in the area of safety as well. The Office of Inspection and Enforcement has not included a safely-related

component in its interpretation of an "unexpected condition." This is inconsistent with the NRC's previous statements that 10 CFR 50.72 notifications were reserved for serious events with an impact on the public health and safety. The I&E interpretation of the regulation in the present instance also ignores the history of industry concern over the definitiveness of notification thresholds.

Therefore, considering the above deficiencies of the 10 CFR 50.72 regulation, the intent of that regulation as stated in NRC formal guidance, and the firm conviction of plant personnel that a "serious event" did not occur, the failure to report the discovery of substantial amounts of water in the containment building until October 20, 1980 could not comprise a Severity Level III violation under the October 7, 1980 Interim Enforcement Policy (45 FR 66754). Among the goals of the Policy is stated to be "enhancing the degree of protection of public health and safety" (Id.). Severity Level III violations are stated to be "violations [which] involve actual or high potential impact on the public" (45 FR at 66755). And in the particular area of reporting requirements, the Policy states that:

> "the severity level of a violation involving the failure to make a required report to the NRC will be based upon the significance of and the circumstances surrounding the matter not reported" (Id.).

The particular basis upon which the Notice of Violation non-reporting Severity Level III claim rests is Supplement I.C.2

of the Interim Enforcement Policy, which describes a Severity Level III violation as:

> "A system designed to prevent or mitigate a serious safety event not being able to perform its intended function under certain conditions (such as not operable unless offsite power is available)." (45 FR at 66758).

Nowhere in the Notice of Violation is it explained which serious safety mitigative or preventive system was unable to perform in the October 17 to 20 period, and in fact our investigations have established that no safety functions were precluded during this period. As noted above, neither fan cooler systems nor occasional leaks from these systems were perceived by the industry or the NRC to present substantial safety degradation possibilities at the time of the leakage event, inasmuch as these systems have not been included in NRC safety-related pronouncements and in the interpretive materials accompanying 10 CFR 50.72.

Under these circumstances, there is no basis under either the NRC rules for reactor operator conduct or the Interim Enforcement Policy for a finding of a Severity Level III violation. Even a Severity IV violation, one rank lower than here claimed, requires either the exceeding of a license limit or a failure "that measurably degrades the safety of operations, incident response, or the environment" (45 FR at 66758). Thus the October leakage event at Indian Point Unit 2 does not even qualify as a Severity IV violation under the new Interim Enforcement Policy.

Giving the factual conclusions of the Office of Inspection and Enforcement's investigation every credence (notwithstanding that its conclusions differ in material respects from the licensee's), under the Interim Enforcement Policy it would also be difficult to claim a Severity V violation for plant conditions from October 17 to 20, such a violation comprising "other violations, such as failure to follow procedures, that have other than minor safety or environmental significance" (Id.).

The unprecedented interpretation of both "unexpected condition" and the Interim Enforcement Policy as contained in the Notice of Violation could thus not have been reasonably anticipated by the licensee. Consolidated Edison accordingly disagrees that a Severity Level III violation of 10 CFR 50.72 occurred in connection with the claimed non-reporting, and denies that there is any appropriate basis for accruing separate violations for each day of licensee misinterpretation.

PLANNED ACTIONS TO PREVENT RECURRENCE:

We are revising our internal procedures to increase the scope and timeliness of event reporting to NRC, and to state and local officials. In the future, to serve what now appears to be NRC's desires, events such as this leakage event will be reported promptly. In addition, we also recognize that it is clearly prudent to assure that event reporting in excess of regulatory requirements is accomplished in certain circumstances. Accordingly, an added objective of our revised internal procedures will be to accomplish this enhanced reporting without placing an excessive or undue burden on licensed operations personnel that would divert them from accomplishing their primary safety functions. These notification procedure changes will be accomplished prior to return to service of the unit.

OFFICE OF INSPECTION AND ENFORCEMENT STATEMENT OF ALLEGED VIOLATION

I.B Technical Specification 6.9.1.7.1 states, in part, that:

"The types of events listed below shall be reported within 24 hours of identification...

c. Abnormal degradation discovered in... primary containment...

Contrary to the above on October 17 and 18, 1980, leaks were discovered in several fan cooler units. These leaks constituted abnormal degradation of primary containment and was not reported to the NRC until October 20, 1980. This violates the 24 hour reporting requirement.

In accordance with Footnote 17 to Section B of the Interim Enforcement Policy this is categorized as a Severity Level III violation.

DISCUSSION:

On October 17 and 18, 1980, plant design features described in the FSAR to monitor and isolate the fan cooler units as intended should they become a source of leakage of radioactivity were operable and the technical specification requirements for containment integrity were satisfied. Therefore, no report pursuant to Technical Specification 6.9.1.7.1.c was submitted or required.

This judgment is supported by I&E Bulletin No. 80-24, issued by the NRC on November 21, 1980, which promulgates new requirements for all licensees resulting from the investigation of the October 17, 1980 event at Indian Point 2. Specifically, item 2.f of that Bulletin now requires that all licensees:

> "[e]stablish procedures to notify the NRC of <u>any</u> [emphasis added] service water system leaks within containment via special licensee event report (24 hours with written report in 14 days) as a degradation of a containment boundary."

Thus, the NRC has determined that service water leaks of the type experienced at Indian Point 2 (or for that matter any leak at all regardless of size) should henceforth be considered and reported as "degradation" (but not abnormal degradation) of a containment boundary. This indicates that a broader interpretation should now be applied to what is reportable under Technical Specification 6.9.1.7.1.c for Indian Point 2 and for all licensees in general.

This requirement was not in effect on October 17, 1980, however, and cannot properly be applied retrospectively in order to sustain a violation against Consolidated Edison. Because the NRC's perception that such leakage should be reported within one day occurred after and as a result of the October 17, 1980 event, it cannot be heard to claim that a report pursuant to Technical Specification 6.9.1.7.1.c was required at the time of the event.

Consolidated Edison accordingly denies that paragraph I.B of the Notice of Violation properly sets forth a violation.

The assumption that the supposed violation of a 24-hour reporting requirement of "abnormal primary containment degradation" constitutes a Severity Level III violation is said to rest on Footnote 17 of the Interim Enforcement Policy (45 FR at 66757), which states that:

> "All violations associated with a particular event or problem will be categorized at the same severity level, even though similar violations, if not associated with the event, might otherwise have been categorized at a lower severity level (e.g., the failure to post a radiation warning sign, which would normally be a Severity Level IV violation, would be categorized as a Severity Level II violation if it contributed to an actual overexposure exceeding 5 rems)."

Thus, this claim of violation associated with a 24-hour reporting requirement may only be of the Severity Level of the one-hour reporting claim alleged under 10 CFR 50.72. As set forth at pp. 7-10, above, neither may properly be considered a Severity Level III violation.

Furthermore, Footnote 17 refers to discrete acts, such as the failure to post a warning sign, which contribute in some causal way to a more serious violation. The characterization given to Footnote 17 here instead impermissibly seeks to create multiple violations from a single act (i.e., the alleged non-reporting), and to create two violations when only one may exist. Given the Interim Enforcement Policy's provisions for escalation of enforcement sanctions for repeat violations (45 FR at 66757-58), the number of violations with which a licensee is charged is of more than academic interest. Even under the Staff's view of the facts, Consolidated Edison has committed but one act of non-reporting, and it is accordingly unfair and inappropriate for it to accumulate two violations for the single alleged omission.

PLANNED ACTIONS TO IMPROVE NOTIFICATION PROCEDURES:

Although we do not agree that this event was reportable as stated, the enhanced reporting described as the corrective action for Item I.A will assure a more comprehensive information flow to the regulatory staff in a manner sufficient to permit independent assessment of the potential consequences of such events. In addition, the requirements of IE Bulletin 80-24 Action Item 2.f. will be followed for reporting leaks in the service water system inside containment. These notification procedure changes will be accomplished prior to return to service of the unit.

OFFICE OF INSPECTION AND ENFORCEMENT STATEMENT OF ALLEGED VIOLATIONS

II. The station Technical Specifications and Quality Assurance Program prescribe the management controls designed to prevent or mitigate a serious safety event. A number of violations of management controls required in these documents occurred. The highest Severity Level associated with these violations is Severity Level III. Because you could reasonably have been expected to have taken effective measures to prevent this occurrence, civil penalties for these violations have been increased by 25%. Therefore, a Civil Penalty - $50,000 is proposed. The civil penalty has been distributed to the separate violations as indicated below:

A. Technical Specification 6.5.1.6 states in part that, "The Station Nuclear Safety Committee shall be responsible for:...

f. Review of facility operations to detect potential safety hazards...."

Contrary to the above, the Station Nuclear Safety Committee did not review, prior to a reactor start-up on October 20, 1980, the potential safety hazards associated with the flooding event of October 17, 1980 during which the hot reactor vessel and various stainless steel components were wetted with cold, brackish river water.

This is a Severity Level III violation (Supplement I.C.2 of the Interim Enforcement Policy). Civil Penalty - $20,000.

DISCUSSION:

Paragraph II of the Notice of Violation concludes as an overall matter that "a number of violations of management controls" had been identified, and that the highest was Severity Level III, representing "violations [which] involve actual or high potential impact on the public" (45 FR at 66755). The "more serious" violation as set forth in the Notice of Violation is II.A, alleging that the Station Nuclear Safety Committee failed to perform a metallurgical safety review prior to reactor startup on October 20, 1980. Each of the remaining five separate violations contained in paragraph II was escalated to Severity Level III via Footnote 17 of the Interim Enforcement Policy, quoted above at p. 14, notwithstanding that most of the alleged violations would be of a lesser severity absent escalation. In order to be subject to escalation under Footnote 17, the violations must all be "associated with a particular event or problem," with the lesser violations "contribut[ing]" to the more serious violation.

Prior to plant startup on October 20, 1980, the accumulation of water on the containment floor was not considered a significant event from a safety standpoint requiring Station Nuclear Safety Comittee review. At that time, it had not been ascertained that the reactor vessel had been wetted, and there was no known potential safety hazard to review.

The theoretical safety issues presented by the wetting of the reactor vessel and the various stainless steel components have upon subsequent review been found not to be safety problems at all. Even though there were no actual safety problems, and even though the Safety Committee was unaware of the vessel wetting at the time of this alleged violation, Consolidated Edison nonetheless agrees that the Committee should have reviewed all relevant safety considerations -- however remote -- after the discovery of substantial amounts of water inside containment and prior to reactor startup. We thus acknowledge that a violation occurred as set forth in paragraph II.A of the Notice of Violation.

Our acknowledgement that the Station Nuclear Safety Committee did not consider whether there was a potential safety hazard is certainly not the same as saying that any hazard ever existed. No safety functions were ever precluded during the leakage event, and as fully borne out by our subsequent investigation, at no time was there any potential or actual hazard. The Company therefore denies that this violation constitutes a Severity Level III violation under the Interim Enforcement Policy, there being no "actual or high potential impact on the public" (45 FR at 66755). Since the paragraph II.A violation cannot be properly characterized as a Severity Level III violation, neither can the "bootstrapped" violations alleged in paragraphs II.B through II.F.

Consolidated Edison also disputes that the Notice of Violation properly states a deficiency in a "system," as that concept is set forth in Supplement I.C.2 of the Interim Enforcement Policy, quoted above at p. 9. It was generally thought prior to this Staff action that "system" contemplated hardware, not personnel, and that presumption is furthered by the Enforcement Policy's reference to "off site power being unavailable" as an illustration.

To the extent that the Policy is to be applied to management, no indication has been provided as to when an isolated instance of personnel error (such as the inconel to stainless weld violation of paragraph II.F) might suddenly be elevated to the status of a serious violation by merely characterizing it as a "system" failure. For example, none of the lesser severity violations alleged in paragraphs II.B through II.F contributed to the paragraph II.A violation as required by Footnote 17, yet all are characterized -- we submit inappropriately -- as Severity Level III because of a supposed connection with a "system" failure.

Lastly, the Company disputes that the management control system violations alleged in paragraph II of the Notice of Violation may properly be increased by 25 percent. Under the Interim Enforcement Policy, such an increase would be appropriate only "...in cases where a licensee could reasonably have been expected to have taken effective

preventive measures" (45 FR at 66756). The Notice of Violation suggests no factual basis for such a finding, and indeed the Interim Enforcement Policy contemplates egregious situations where a licensee disregards actual knowledge of a condition gained from prior NRC inspection or licensee audits and the like (45 FR at 66756).

On the specific subject of the adequacy of our management control system at Indian Point 2 as raised in paragraph II of the Notice, we refer to an audit conducted by the Office of Inspection and Enforcement in their report dated September 2, 1980. The report of that audit found that:

> "It was the general view that management interpreted the technical specifications literally or conservatively (with respect to safety) and that plant operators had no reservations about shutting the plant down if, in their opinion, technical specification limits or other safety considerations required it. Based on the above interviews, indications were that plant and corporate management's first priority was safe operation and that the operating staff was aware of this priority."

Violations premised upon an overall failure in the management control system should accordingly be dismissed.

PLANNED ACTIONS TO MODIFY SNSC RESPONSIBILITIES:

Three areas of activity are planned to increase SNSC awareness of situations where its review may be desirable, and to encourage improvement in its overall effectiveness:

1. The role of SNSC, especially in the area of review of facility operations to detect potential safety hazards, will be re-emphasized to committee members as well as all members of the operations staff.

2. The training and periodic retraining programs will be enhanced by revision to appropriate curriculum material to emphasize the safety review role of SNSC and of individuals, and to increase the awareness of operations staff to situations where SNSC review may be desirable.

3. SNSC will participate in a systematic review, on a periodic basis, of maintenance activities with potential safety consequences even though they may not be associated with a change or modification to the facility otherwise requiring review.

4. Operating procedures will be revised to require prior SNSC review of reactor startup if the Senior Watch Supervisor or Shift Technical Advisor either have not positively identified the cause of the reactor trip

or have determined that startup may involve unusual conditions.

These changes will be accomplished prior to return to service.

In addition, Consolidated Edison will review the requirements imposed on its safety committees with an aim towards reducing the volume of routine documentation review and increasing the intensity of safety matter reviews. This is an ongoing activity that will increase the ability of its various Committees to also review routine operations objectively. Progress will be reviewed by Consolidated Edison management annually.

OFFICE OF INSPECTION AND ENFORCEMENT STATEMENT OF ALLEGED VIOLATION

II. B. Technical Specification 6.8.1 requires that procedures shall be established, implemented and maintained to meet the requirements and recommendations of Appendix A to Regulatory Guide 1.33-1972, and ANSI N18.7-1972, sections 5.1 and 5.3.

1. Regulatory Guide 1.33-1972, Appendix A, paragraph H.1, calls for procedures of a type appropriate to the circumstances to assure that instruments and controls are properly calibrated and adjusted to maintain accuracy.

2. Regulatory Guide 1.33, Appendix A, paragraph H.2 calls for procedures to implement each surveillance test, inspection or calibration listed in the Technical Specifications. Technical Specification 3.1.F.1 requires a safety evaluation whenever reactor coolant system leakage is indicated by the means available.

3. ANSI N18.7-1972, Section 5.3, states that procedures shall provide an approved preplanned method of conducting operations. Section 5.3.2.6 states that limitations on parameters being controlled and appropriate corrective measures to return the parameter to the normal control band should be specified.

4. ANSI N18.7-1972, Section 5.1.6.1, states that maintenance or modifications that may affect functioning of safety related systems shall be performed to assure quality and that maintenance shall be properly preplanned and performed in accordance with written procedures appropriate to the circumstances.

Contrary to the above, procedures were not established, implemented and maintained in that, respectively:

1. No setpoints for containment sump pump operation were included in the surveillance test, PT-R2A, "Containment Sump Level Analog Test", Revision 2, which verified sump pump operability; and,

2. Procedures were not established or implemented for the condensate flow leak detection system or the containment humidity detectors which would satisfactorily implement Technical Specification 3.1.F.1 to detect reactor coolant system leakage; and,

3. Procedures were not established which would provide for a preplanned method of controlling the containment sump level. Specifically, no control band or maximum sump level was specified, nor were corrective measures detailed; and,

4. Site administrative procedures were not established, implemented and maintained to provide guidance as to when written approved procedures were required for maintenance activities or as to when maintenance activities would constitute a modification, both of which require review and concurrence by the Station Nuclear Safety Committee.

In accordance with Footnote 17 to Section B of the Interim Enforcement Policy this is categorized as a Severity Level III. Civil Penalty - $10,000.

DISCUSSION OF PARAGRAPH II.B.1 AND 3:

Because of the similarity of the allegations in these two paragraphs, they are covered by the same discussion. ANSI N18.7 - 1972 states in Section 1. that the requirements of this Standard apply to all activities affecting the safety-related functions of nuclear power plant structures, systems, and components. In the design basis for the plant, the containment sump pumps are not defined as safety related by either the Commission or Consolidated Edison. Accordingly, operability and surveillance requirments for these pumps are not included in the facility Technical Specifications.

Float settings for starting and stopping the pumps were set so as not to allow the level in the sump to reach the 46' elevation. No procedures were deemed necessary to document the on/off setpoints. Based on operating experience prior to commercial operation, the original settings for starting and stopping the containment sump pumps were changed to permit the pump, once started, to run for a longer period to prevent frequent starts and stops. The setting criteria used were to make certain the pump did not run dry and that the sump was approximately one-half full before the pump started. Test Procedure PT-R2A is used to verify this during refueling outages. Thus a procedure did exist, appropriate to the circumstances, for the containment sump pumps.

Successful operation of these pumps since initial installation supports the appropriateness of the procedures.

It should be noted that operation of these pumps using the above criteria for float setpoints had been satisfactory since the start of commercial operation of the unit up until the occurrence. Furthermore, the absence of a float setpoint procedure did not contribute to the accumulation of water in the containment during October 1980.

Contrary to the allegation of paragraph II.B.3, a procedure was established to provide a preplanned method of controlling the containment sump level. Level control switch settings were physically preset to assure a proper control band and procedure PT-R2A was implemented to verify operability.

As stated previously, no safety significance was accorded the containment sump pumps, therefore a setpoint calibration value was not specifically included in a procedure nor considered necessary. The absence of a specific setpoint value did not contribute to the accumulation of water in the containment during October, 1980. Consolidated Edison accordingly denies that a violation is set forth in paragraphs II.B.1 or 3 of the Notice of Violation.

For the reasons set forth at pp. 7-10 and 19-20 above, these matters are in any event not properly Severity Level III violations. In paragraphs II.B.1 and 3 of the Notice of Violation there is additionally an inappropriate and unwarranted

application of Footnote 17 of the Interim Enforcement Policy, whereby contentions relating to sump pump operations and procedures are "bootstrapped" into a claimed Level III status, based upon their purported similarity with a distinct claim of violation relating to the Station Nuclear Safety Committee, paragraph II.A. This conflicts with the requirements of Footnote 17 that "bootstrapped" violations in such situations must be "associated with a particular problem."

DISCUSSION OF PARAGRAPH II.B.2:

Regulatory Guide 1.33 Appendix A, paragraph H.2 calls for procedures to implement each surveillance test, inspection and calibration listed in the Technical Specifications. Requirements for surveillance testing, inspection and calibration are contained in Section 4 of the Indian Point Unit 2 Technical Specifications. There are no requirements in Section 4 of the Unit 2 Technical Specifications for periodic surveillance testing, inspection and calibration of the Dew Point and Fan Cooler Condensate Flow Monitoring Systems. However, to assure operability of these systems, they are included in a biennial surveillance program. In accordance with this program, the dew point recorder and weir level transmitters are calibration checked every two years and readjusted for proper operation where necessary.

Additionally, there is a procedure for evaluating reactor coolant system leakage. The procedure, Station Operating Procedure 1.7 ("Reactor Coolant System Leakage Surveillance and Safety Evaluation"), describes the methods and monitoring systems available for determining Reactor Coolant System leakage rates. The procedure describes how the Volume Control Tank level, the Containment Air Particulate and Radiogas activity monitors, the Dew Point System and the Fan Cooler Condensate Flow Monitors are used in determining the Reactor Coolant System leakage rate.

The Technical Specifications require two reactor coolant leak detection systems employing different principles be operable when the reactor is critical. One of these systems must be sensitive to radioactivity. The humidity and weir flow systems are back-ups to the radiation detectors and, in fact, only one of these backup systems need be operable according to the Technical Specification requirements. The procedure requires that changes in the readings of these systems, as well as their absolute value, be taken into consideration when evaluating the data. The procedure also points out that if dew point and/or weir flow values increase, but changes are not observed in any of the radiation detection systems, then the leak most probably is not from the Reactor Coolant System.

These procedures adequately implement the Technical Specifications applicable to Reactor Coolant System leakage. We accordingly do not concur that procedures were not established or implemented for the condensate flow leak detection system or the containment humidity detectors which would satisfactorily implement Technical Specification 3.1.F.1. Furthermore, for the reasons set forth at pp. 7-10 above, the claimed violation would not under any circumstance be properly considered as Severity Level III. No suggestion is made that the claimed noncompliance with condensate flow leak detection system requirements "involve(s) actual or high

potential impact on the public" (45 FR 66755), and this supposed violation is unrelated to the Station Nuclear Safety Committee contention to which it is bootstrapped. Neither claim is properly a Severity Level III violation, see pp. 18,28.

DISCUSSION OF PARAGRAPH II.B.4:

When the proposed activity has been deemed to be a modification by engineers in the maintenance group, the corresponding "Request for Safety Evaluation" for that activity is processed by the Nuclear Systems Engineering Subsection in accordance with Company procedures.

This Subsection reviews the activity to be performed and determines if the activity falls within the scope of NRC regulations concerning changes. If it does not, a notification to this effect is issued. If it does, a safety evaluation is performed as required by 10 CFR 50.59 to determine whether an unreviewed safety question exists. If no unreviewed safety question exists, the written safety evaluation is issued.

In addition to a safety evaluation, Engineering prepares a modification procedure in accordance with Company procedures. This modification procedure (which includes the safety evaluation) is reviewed by Quality Assurance and the Station Nuclear Safety Committee (SNSC). All safety evaluations are also reviewed by the Nuclear Facilities Safety Committee. If it is determined as a result of these reviews that an unreviewed safety question does exist, and the scope of the activity cannot be changed to preclude such a determination, the activity is submitted to the NRC for review and approval.

Plant maintenance personnel are aware of the requirement to have plant modifications reviewed by the Station Nuclear Safety Committee, and SNSC has in fact reviewed over the years innumerable maintenance activities which constituted plant modifications. The Central QA&R field office at the plant, which reports offsite to the Director of QA&R, reviews and monitors maintenance and modification activities at the plant to assure that the QA program is being implemented.

With regard to guidance as to when maintenance activities constitute a modification, SAO 102 states that:

2.5 Each Subsection or Staff Head charged with the responsibility for development and approval of procedures (of the type described in Reg. Guide 1.33) shall:

2.5.4 Assure that each procedure/procedure change which would render Final Safety Analysis Report or subsequent safety analysis report inaccurate or may involve an unreviewed nuclear safety question is approved, before implementation, by the SNSC.

2.5.5 Assure that each procedure/procedure change which would make the Final Safety Analysis Report, or any later safety analysis report, inaccurate has a written justification of why an unreviewed nuclear safety question is not involved.

SAO 102, Revision 5 dated June 20, 1979 provides guidance as to when written and approved procedures are required for maintenance activities. Specifically, this document states:

2.1 Each proposed procedure/procedure change involving nuclear safety related components and/or operation of same shall receive a pre-implementation review by the Station Nuclear

Safety Committee (SNSC) except in an emergency situation. The attached cover sheet (Exhibit 1) shall document this and other required approval. "Safety related components" are those which are used in activities listed in NRC Regulatory Guide 1.33, Appendix A and:

a. Form or are within the primary pressure boundary or

b. are accident preventing structures and/or systems, or

c. are accident mitigating structures and/or systems, or

d. contain radioactive materials as specified by Operating License DPR 26.

An "emergency", as used herein, exists when continued or safe plant operation is in jeopardy directly due to lack of a procedure/procedure change and SNSC review cannot be readily obtained (e.g., off-hours).

2.5 Each Subsection or Staff Head charged with the responsibility for development and approval of procedures (of the type described in Reg. Guide 1.33) shall:

2.5.1 Have procedure(s) covering activities under his jurisdiction that affect nuclear safety (ANSI N18.7-1976, Par. 5.3).

2.5.3 Assure that each procedure/procedure change involving nuclear safety related components and/or operation of same receives a pre-implementation review by the SNSC except in an emergency situation.

Maintenance Administrative Directive, MAD-4, Rev. 1, dated August 1, 1978, implements the requirements of this SAO with respect to maintenance activities.

We therefore disagree that site administrative procedures were not established, implemented and maintained to provide

guidance as to when written approved procedures were required for maintenance activities, or when maintenance activities would constitute a modification, both of which required review and concurrence by the Station Nuclear Safety Committee. Accordingly, Consolidated Edison denies that a violation is stated in paragraph II.B.4 of the Notice of Violation.

For the reasons set forth at pp. 7-10 and 18 above, such a claimed violation is not under any circumstances appropriately construed as Severity Level III.

PLANNED ACTIONS TO MODIFY MAINTENANCE AND SURVEILLANCE PROGRAM:

While Consolidated Edison disagrees that any violations occurred as claimed in paragraph II.B of the Notice of Violation, the following steps are being taken as a result of this event:

1. The containment sump system will be reclassified "Class A" in accordance with Con Edison's CI240-1 and the equivalent level of quality required by 10 CFR 50 App. B. Thus all future maintenance and modification work on this system will be in accordance with these requirements.
2. Procedures for testing, calibration, and maintenance will be developed and implemented consistent with the new classification.
3. Consolidated Edison will proposed revisions to the Facility Technical Specifications to include limiting conditions for operation and surveillance requirements for the various leakage detection systems.
4. Revisions to key administrative directives will be made to clarify the guidance on what constitutes a modification.
5. An extensive review is being conducted of the Nuclear and Non-Nuclear equipment surveillance and testing programs with the goal of upgrading the Non-Nuclear

programs to the same level of attention given safety-related nuclear systems.

Item 1 will be accomplished immediately and Items 2, 3, and 4 will be completed prior to return to service.

OFFICE OF INSPECTION AND ENFORCEMENT STATEMENT OF ALLEGED VIOLATION

II. C. 10CFR 50, Appendix B, Criterion II requires that:

"...The quality assurance program shall provide control over activities affecting the quality of the identified..systems, and components...".

FSAR Volume A, Attachment A-2, "Quality Assurance Program (ANSI N18.7 Format) Revised June, 1977", Foreward, states that:

"The following quality assurance program conforms to the requirements of 10 CFR 50, Appendix B. Additionally, Con Edison commits to have a Quality Assurance Program satisfying the requirements and guidelines of the following ANSI Standards and complying with the Regulatory Position in the Regulatory Guides as modified by Table A and Table B.

ANSI Standards

N18.7-1976 'Administrative Control and Quality Assurance for the Operational Phase of Nuclear Power Plants'."

ANSI 18.7, Paragraph 5.2.7.1, "Maintenance Programs" states that:

"The causes of malfunction shall be promptly determined, evaluated and recorded..."

Contrary to the above, despite continued malfunctions (i.e., leaks) in the fan cooler units between 1973 and October, 1980, the causes of the malfunctions had not been determined or recorded, and evaluations of the causes had not been completed.

In accordance with Footnote 17 to Section B of the Interim Enforcement Policy this is categorized as a Severity Level III Violation. Civil Penalty - $10,000.

DISCUSSION:

NRC regulations, 10 CFR, Part 50, Appendix B, Criterion II, require that:

> The quality assurance program shall provide control over activities affecting the quality of the identified structures, system, and components, to an extent consistent with their importance to safety.

More specifically with respect to malfunctions, NRC regulations, 10 CFR, Part 50, Appendix B, Criterion XVI, states that:

> Measures shall be established to assure that conditions adverse to quality such as malfunctions... are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition.

The NRC regulations indicate that the extent of control over plant systems should be consistent with the importance of that system to safety. The applicable regulations require that all malfunctions be "promptly identified and corrected".

Consolidated Edison's Quality Assurance (QA) Program complies with the NRC regulations and has been approved by the NRC. We accordingly do not agree that a violation occurred as claimed in paragraph II.C of the Notice of Violation. All leaks in the fan cooler units were "promptly identified and corrected". The leaks were all pin hole leaks in brazed or welded joints, or in fan cooler tubes.

In addition to identifying and repairing all leaks, the causes of the fan cooler leaks were in fact both determined and, to the extent appropriate, documented. In 1973, leaks developed in the fan cooler unit motor cooler heat exchangers. The leaks developed in the tube stub ends where they were brazed to the manifold. All five motor coolers were removed and sent to Consolidated Edison's laboratory. Careful inspection by Consolidated Edison revealed evidence of some instance of poor fit-up in the brazing operation during manufacturing, with non-uniform clearance and non-uniform flow of brazing material. This laboratory analysis was documented in a March 1973 report of the analysis, which was widely distributed within Consolidated Edison, and in particular to Indian Point plant personnel. The design and fabrication of the motor cooler stub ends were very similar to the design and fabrication of the fan cooler main heat exchanger stub ends. Therefore, it was a reasonable engineering judgment to assume that the few subsequent leaks in the stub ends on both the motor cooler and the main heat exchangers were also caused by poor control of the brazing operation. This judgment has been confirmed by our examinations and tests of the heat exchangers since October 17, 1980.

Prior to the October, 1980 incident, all leaks in the mid body tubes developed only in one section of fan cooler unit 25. After the first leaks developed in the tubes of that

section, Consolidated Edison promptly ordered and obtained a replacement section.

The leaks in the service water lines were in welded joints in the cement lined carbon steel pipe, in close proximity to the copper nickel piping. These leaks are known to be caused by corrosion accelerated by galvanic corrosion due to the proximity of the dissimilar metals.

NRC Staff's assertion that the evaluation of the cause of leaks had not been completed is true. In order to confirm prior evaluations, monitoring of the leakage continued during operation to assure that leaks were not caused by processes not previously detected.

These evaluations continued even after the decision to replace the fan cooler units was made, confirming that Consolidated Edison had correctly identified the cause of the leakage, as described above.

For these reasons, there was no violation of applicable NRC regulations in connection with the prompt identification and correction of the causes of fan cooler leakage prior to October 1980. Consolidated Edison's attention to the fan cooler system and its maintenance was entirely appropriate to the operational experience. Even if a noncompliance be assumed, arguendo, for the reasons set forth at pp. 7-10 and 18 above, a Severity Level III violation would not be present, as alleged in Paragraph II.C of the Notice of Violation.

PLANNED ACTIONS TO MODIFY QUALITY ASSURANCE PROGRAM:

The licensee intends to modify its conduct of quality assurance activities so that they are in excess of present regulatory requirements. The Station Nuclear Safety Committee and the Quality Assurance and Reliability Department will participate in systematic reviews on a periodic basis of equipment malfunctions and their repairs. Action to initiate these reviews will be taken prior to return to service.

OFFICE OF INSPECTION AND ENFORCEMENT STATEMENT OF ALLEGED VIOLATION

II. D. 10 CFR 50, Appendix B, Criterion II, states "...The quality assurance program shall provide control over activities affecting the quality of the identified... systems, and components..."

FSAR Volume A, Attachment A-2, "Quality Assurance Program (ANSI N18.7 Format) Revised June, 1977", Foreward, states "The following quality assurance program conforms to the requirements of 10 CFR 50, Appendix B. Additionally, Con Edison commits to have a Quality Assurance Program satisfying the requirements and guidelines of the following ANSI Standards...

ANSI Standards

N18.7-1976 'Administrative Control and Quality Assurance for the Operational Phase of Nuclear Power Plants'."

ANSI 18.7-1976, Paragraph 5.2.7.1, Maintenance Programs, states in part. "A maintenance program shall be developed to maintain safety related...systems...at the quality required for them to perform their intended functions...Planning for maintenance shall include evaluation of the use of...materials in the performance of the task...".

10 CFR 50.59(b) states, in part, that the licensee shall maintain records of changes in the facility which include a written safety evaluation that provides the bases for the determination that a change does not involve an unreviewed safety question.

Technical Specification 6.5.1.6 requires that "The Station Nuclear Safety Committee (SNSC) shall be responsible for: ...

d. Review of all proposed changes or modifications to plant systems of equipment that affect nuclear safety..."

Contrary to the above, modifications were made to the fan cooler unit cooling coils and service water lines during maintenance performed between 1973 and July, 1979 without review by the SNSC and without an evaluation being conducted to demonstrate the suitability

of epoxy sealant material to perform its intended function under loss of coolant accident (LOCA) conditions. In August, 1979 an evaluation of the epoxy sealant material was made, which did not consider all of the post-LOCA conditions or the specific mode in which the sealant was used. Subsequent to this, the plant was operated at power and additional repairs were made on July 7 and 25, 1980 and on October 3, 18 and 19, 1980.

In accordance with Footnote 17 to Section B of the Interim Enforcement Policy this is categorized as a Severity Level III Violation. Civil Penalty - $5,000.

DISCUSSION:

Since the use of epoxy in no way alters the basic structure of the piece of equipment on which it is applied and does not affect or degrade the equipment's ability to perform its function, the use of epoxy was correctly determined to be a repair and not a modification. There was accordingly no non-compliance with either ANSI or NRC requirements respecting system changes or modifications. As with most maintenance or repair activities, an epoxy repair simply returns the equipment to its original leak tight condition without changing its safety function. There was no change to the system as described in the Final Safety Analysis Report. All past changes to the fan cooler service water piping that were deemed modifications were reviewed by the Station Nuclear Safety Committee. Therefore, since the use of epoxy is a repair, rather than a modification or a change, there is no requirement that its use be reviewed by the Station Nuclear Safety Committee.

At the time that Con Edison used epoxy in maintenance of the fan cooler units, the ability of epoxy to perform satisfactorily during normal operation and under conditions of high temperature, pressure, radiation, and caustic solutions was known and documented. The adequacy of these repairs has been proven in practice, because no epoxy repair has ever failed after being placed in service on the Indian Point Unit 2 fan cooler system.

Repair of leaks using epoxy has proven to be the most viable method of repair as evidenced by test data and actual field experience. It appears to be more satisfactory than brazing in the repair of stub tube leaks from a leak-tightness viewpoint and totally eliminates the risk of weakening adjacent brazed areas. Epoxy has also proven to be the most reliable means of repair for main tube leaks, greatly reducing the possibility of damage to adjacent tubes, as compared to brazing, for example. This same material is also considered to be a reliable means of repair for brazed field joints to avoid the header distortions which could occur by braze repairs. Use of epoxy on weld leaks in cement lined piping avoids the risk of damaging the adjacent cement liner. This material has, in fact, stood up well during the past summer and winter river water temperature cycles when used for repairs at Indian Point.

With respect to accident conditions, the acceptance criteria were that the repairs maintain integrity under conditions of radiation, humidity, containment pressure and temperature and water temperature conditions that might result from a loss of coolant accident. These consist of an accumulated dose of 2×10^8 rads, with relative humidity conditions initially at 100% and containment air initially at 271 F and 47 psig.

One type of epoxy used was supplied by Bonded Products, Inc. Our review of the Handbook of Epoxy Resins (1967) and discussions with the epoxy supplier indicated that the aromatic type epoxy was among the types of epoxies tested for radiation under a U.S. Government research contract and for Atomics International. This type of epoxy has an exposure capability of 1×10^9 rads before the point of incipient damage is reached, and provides a safety margin of a factor of 5 or more over the accumulated dose expected for the full course of a loss of coolant accident. Thus, the epoxy has been established as able to perform its function during an accident without risk of radiation damage.

The Bonded Products, Inc. epoxy is specified by the manufacturer to meet the requirements of military specification MIL-R-17882D dated June 30, 1978 and the earlier version of this document dated March 6, 1961, for general use on structures, piping and equipment including copper nickel pipe and tubing. This military specification includes patches for pressures of 200 psi minimum on copper nickel tubing. For Indian Point 2, actual and potential service conditions for cooler repairs are well below this specification pressure differential. Additionally, the manufacturer's test program has qualified this epoxy for continuous service at 280 F.

A confirmation evaluation was performed during August 1979 at

Consolidated Edison's Astoria Chemical Laboratory on 90-10 copper nickel tubing. After curing a sample tube with epoxy for 24 hours at 80 F and at 120 F, the test was performed in a steam bath at 212 F for 24 hours. The 212 F test temperature is above the maximum calculated temperature that the epoxy-to-tube interface would actually reach during LOCA conditions. After removal from the steam bath, an attempt was made to chip the epoxy from the tube and the epoxy was examined for degradation. The epoxy adhered to the material and there was no indication of spalling and no evidence of degradation. This further substantiated that the material was suitable for repairs of the containment coolers and motor coolers.

Additionally, during November 1980, NUS Corporation tested Bonded Products, Inc. epoxy repairs on actual tube samples removed from the Indian Point 2 cooler units. These tests included cycling between 32 F and temperatures in excess of 271 F using ice water, steam and condensate conditions at NUS Corporation's facilities. The test samples had repairs made on 1/8 inch drilled holes. The repairs successfully withstood 140 psig tests reconfirming the original evaluation.

Another type of epoxy, which has been used since 1979, was supplied by Masterbond. This type of epoxy qualifies to General Services Administration Federal Specification numbers MMM A132 and A 134 which in part require operating temperature

capability to withstand a 300 F continuous operating temperature. The Handbook of Epoxy Resins also reports the well known moisture and temperature capabilities of this epoxy. In addition, radiation exposure tests performed by both Ciba-Geigy and Radiation Dynamics (1973) have qualified this epoxy to 1×10^9 rads. Furthermore, this type epoxy is used by Wyle Laboratories as a conduit sealer during the performance of various environmental qaulifiication testing and it is specifically exposed to environments of 340 F and 65 psig. Accordingly, the Masterbond epoxy more than satisfies the Indian Point 2 accident criteria and has been acceptable for use.

For the foregoing reasons, Consolidated Edison does not agree with the violation set forth in paragraph II.D of the Notice of Violation. Even if a violation were deemed to have occurred, for the reasons set forth at pp. 7-10 and 18 above, such a violation would not properly be categorized as Severity Level III.

PLANNED ACTIONS RESPECTING MAINTENANCE AND REPAIR ACTIVITIES:

At the direction of the Chairman of the Nuclear Facilities Safety Committee, the Quality Assurance and Reliability Department is developing a new program to assure that a systematic review for safety implications is conducted of maintenance activities. This program will exceed present regulatory requirements applicable to maintenance and repairs, and will be initiated prior to return to service.

OFFICE OF INSPECTION AND ENFORCEMENT STATEMENT OF ALLEGED VIOLATIONS

II. E. 10 CFR 50, Appendix B, Criterion XVI requires that "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deviations, defective material and equipment, and nonconformances are promptly identified and corrected."

FSAR Volume A, Attachment A-2, "Quality Assurance Program (ANSI N18.7 Format) Revised June, 1977", Section 5.2.11, "Corrective Actions", states "Measures have been established which ensure that conditions adverse to plant safety which may occur during work, e.g., maintenance, are promptly identified in a Quality Control Inspection Report (QCIR) or a Deficiency Report (DR) and corrected.... The action addressee on the Quality Control Inspection Report (QCIR)... is responsible for either correcting the nonconformance or designating the organization responsible for completing the necessary corrective actions. The managements of these designated organizations are responsible for taking the necessary corrective actions." Implementing Procedure SAO-113, Quality Control Reports and Stop Work Authority, Revisions 0 and 1, Paragraph 2.7, states in part, "In any case where the recipient of a QCIR is unable to make a schedule... or does not agree with the specific action called for, he will so inform the ... QA Engineer in writing. Feedback to the QA Engineer per the requirements above should be provided promptly, i.e., generally within three (3) working days of the QCIR receipt."

Contrary to the above, the measures established did not assure prompt correction in that:

1. The following QCIRs had not been responded to promptly as no response has been received as of October 29, 1980.

   -- 79-2-14, issued April 2, 1979
   -- 79-2-27, issued May 27, 1979
   -- 79-2-43, issued July 17, 1979
   -- 79-2-44, issued July 20, 1979
   -- 79-2-74, issued September 17, 1979
   -- 80-2-17, issued February 16, 1980
   -- 80-2-19, issued March 17, 1980
   -- 80-2-33, issued September 4, 1980

2. The following QCIRs were closed by the Quality Assurance Engineer based on various types of followup action but had never been responded to in writing.

   -- 78-2-27, issued February 23, 1978
   -- 79-2-66, issued August 27, 1979
   -- 79-2-77, issued November 29, 1979
   -- 79-2-75, issued September 20, 1979
   -- 80-2-13, issued February 14, 1980
   -- 80-2-28, issued July 25, 1980
   -- 80-2-29, issued July 25, 1980
   -- 80-2-39, issued October 2, 1980

3. The following QCIRs which are closed had not been responded to promptly.

   -- 73-2-184, issued November 15, 1973; responded to May 5, 1974
   -- 76-2-001, issued January 19, 1976; responded t March 9, 1976
   -- 77-2-89, issued June 9, 1977; responded to August 3, 1977
   -- 80-2-25, issued May 13, 1980; responded to July 17, 1980

In accordance with Footnote 17 to Section B of the Interim Enforcement Policy this is categorized as a Severity Level III Violation. Civil Penalty - $5,000.

DISCUSSION:

Consolidated Edison Station Administrative Order 113 recommends but does not require that a response to a QCIR be filed within three days. This is because in many cases it is impractical to make the appropriate inquiries, evaluate the underlying problems, formulate an approach to corrective action, and respond within that time period. There are no NRC regulations or guidelines which stipulate a particular response time for quality control documents such as QCIRs, and in this regard there were no violations as claimed in the Notice of Violation.

In early 1980, Consolidated Edison engaged a consultant to conduct an audit of our QCIR program. The final report of the audit was received and changes have been initiated to the Corporate Quality Assurance Program as a result of the audit recommendations. These changes will result in upgrading the QCIR system and will address in particular timely responses to QCIRs.

These QCIRs which were closed by the QA Engineer but not formally responded to in writing were used as allowed as an administrative method to track repairs rather than for identifying and correcting conditions adverse to quailty. These QCIRs were appropriately closed out by the QA Engineer on the basis of work plans and activities which he believed would constitute final action related to these repairs.

Con Edison therefore denies that a violation is set forth in paragraph II.E of the Notice of Violation. For the reasons set forth at pp. 7-10 and 18-20 above, the claimed violation could not under any circumstances properly be construed as Severity Level III. The matters referred to in paragraph II.E are not associated with any particular event or problem referred to elsewhere in the Notice of Violation, and did not "contribute" to the Station Nuclear Safety Committee violation (paragraph II.A) to which paragraph II.F is "bootstrapped" in order to conclude that it is a Severity Level III. In fact, the contentions of paragraph II.E., even if a violation were sustained, would only be of a much lesser Severity Level.

PLANNED ACTIONS RESPECTING QUALITY ASSURANCE PROGRAM:

As mentioned previously, appropriate administrative documents related to the Quality Assurance Program are being revised to accomplish certain improvements recommended by our internal audit. These improvements include stipulation of response time for particular non-conformance reports by designated personnel, contacting management of action addressee if response dates are not initially met, and escalation to significant non-conformance report status if response dates are again not met. Periodically, reports of significant non-conformances will be distributed to Vice Presidents of affected organizations, and the Chairman of the Nuclear Facilities Safety Committee. These changes will be made and fully implemented prior to return to service.

OFFICE OF INSPECTION AND ENFORCEMENT STATEMENT OF ALLEGED VIOLATION

II. F 10 CFR 50, Appendix B, Criterion VIII, "Identification and Control of Materials, Parts, and Components", states that:

"Measures shall be established for the identification and control of materials, parts, and components, including partially fabricated assemblies. These measures shall assure that identification of the item is maintained by heat number, part number, serial number, or other appropriate means, either on the item or on records traceable to the item, as required throughout fabrication, erection, installation, and use of the item. These identification and control measures shall be designed to prevent the use of incorrect or defective material, parts, and components."

FSAR Volume A, Attachment A-2, "Quality Assurance Program (ANSI N18.7 Format) Revised June, 1977", Foreword, states that "The following quality assurance program conforms to the requirements of 10 CFR 50, Appendix B. Additionally, Con Edison commits to have a Quality Assurance Program satisfying the requirements and guidelines of the following ANSI Standards...

ANSI Standards

N18.7-1976 'Administrative Control and Quality Assurance for the Operational Phase of Nuclear Power Plants'."

ANSI 18.7-1976, paragraph 5.2.7 states that:

"Maintenance or modifications which may affect functioning of safety-related structures, systems, or components shall be performed in a manner to ensure quality at least equivalent to that specified in original design bases.... Maintenance or modification of equipment shall be preplanned and performed in accordance with written procedures, documented instructions or drawings appropriate to the circumstances which conform to applicable codes."

Contrary to the above, maintenance repairs on the fan cooler unit water heat exchanger flexible hoses were not conducted in a preplanned manner and did not provide for the control and identification of materials in that: MWR 4156 and MWR 6508 completed in 1976 failed to identify the as installed flexible hoses as Inconel 625 per Addendum No. 1 (dated September 2, 1972) to Specificiation 9321-01-248-76, assumed the materials to be austenitic stainless

steel, removed the center section of the existing hose leaving a short 2 inch stub section of the original hose and installed a stainless steel replacement. A P8 to P8, austenitic stainless steel welding procedure was utilized for the P8 to Inconel dissimilar metal joint. An austenitic stainless steel flexible hose was substituted for the Inconel 625 hose required by the design specification.

In accordance with Footnote 17 to Section B of the Interim Enforcement Policy this is categorized as a Severity Level III.

DISCUSSION:

In 1975, Consolidated Edison replaced an Inconel hose having a braided stainless sheath. The then existing hose was believed to be stainless. The replacement hose was 321 stainless steel with 304 stainless steel weld ends. Stub ends of the actual inconel hose were left in place. A weld procedure which was qualified for stainless welding was used because of the incorrect material identification. The two alloys are similar in that they are materials containing chromium and nickel, although inconel contains more nickel than stainless steel, they are both structurally austenitic. Both alloys are suitable for use in the Service Water System.

However, since there was an incorrect material identification, there was a violation as set forth in paragraph II.F of the Notice of Violation.

An Engineering evaluation has been completed which concluded that the welding procedure that was used would yield a weld with an acceptable level of quality, and the examination of the weld conformed to system requirements. These welds have given satisfactory service since 1975.

While this is a violation, it should not be considered a Severity Level III violation under the Policy. There was no exceedence of a Limiting Condition for Operation, and the system was fully capable of performing its intended function.

The materials and weld procedure used are qualified for use in the service water piping system. At worst, this violation could be considered to have "minor safety significance", in which case the violation would be considered a Severity Level VI violation (45 FR at 66758).

This item of noncompliance should not be bootstrapped to a Severity Level III violation merely because it was detected during an investigation in which the NRC Staff asserts there were other Severity Level III violations. This is an inappropriate use of Footnote 17 of the Interim Enforcement Policy, which contemplates an event which contributes in some causal manner to the more serious violation with which it is connected, see pp. 18, 28 above. Here there is no causative connection whatsoever, and the violation is unrelated to the accumulation of water in containment in October 1980.

ACTIONS TO PREVENT RECURRENCE:

This is an isolated instance of an inadvertent misidentification of two very similar materials which has not resulted in any failure for over five years. Maintenance personnel will be cautioned regarding the importance of proper material identification, especially in maintenance operations involving welding. Where questions may exist regarding material identification, metallurgical assistance is to be requested from Engineering and/or the Chemical Laboratories.

The above will be carried out immediately.

OFFICE OF INSPECTION AND ENFORCEMENT STATEMENT OF ALLEGED VIOLATION

III. NRC's Confirmatory Order to Consolidated Edison Company of New York, Inc., dated February 11, 1980, ordered the licensee to establish and man the Shift Technical Advisor (STA) position within ninety days.

NRC's letter to All Operating Nuclear Power Plants, dated September 13, 1979, titled "Followup Actions Resulting From The NRC Staff Reviews Regarding The Three Mile Island Unit 2 Accident," stated that licensees should establish the Shift Technical Advisor position by January 1, 1980, and that "...in order to provide both perspective in assessment of plant conditions and dedication to the safety of the plant, this function (Accident Assessment Function) should have a clear measure of independence from duties associated with the commercial operation of the plant."

A. NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," states: "...that additional technical and analytical capability, dedicated to concern for the safety of the plant, needs to be provided in the control room to support the disagnosis of off-normal events and to advise the shift supervisor on actions to terminate or mitigate the consequences of such events..."; that the position of Shift Technical Advisor (STA) be established to fulfill this function; and that "...when assigned as shift technical advisor, these personnel are to have no duties or responsibilities for manipulation of controls or command of operations."

During the investigation, from October 22, 1980 to November 21, 1980, the NRC interviewed STAs who performed duties during the period from 11:00 p.m. on October 16, 1980 to 07:00 a.m. on October 20, 1980. The STA, stated that, contrary to the above, they are not always called to the Control Room when problems are identified and that operations personnel utilize STAs for routine activities not involving engineering review or evaluation of plant safety, once the plant is shut down.

Also the STAs, on their shift, had not evaluated the propriety of a return to power when it occurred twice on October 17, 1980 and once on October 20, 1980, nor did they evaluate the potential significance of the degraded plant conditions involving leakage from the fan cooler units, wetting of the reactor vessel with cold brackish river water and steam exiting from the instrument thimble holes.

This is a Severity Level III violation (Supplement 1.C.2 of the Interim Enforcement Policy) Civil Penalty - $30,000. The civil penalty of $40,000 for Severity Level III violation has been distributed between this item of noncompliance and the following one, both of which together comprise an event.

B. NRC's letter to All Operating Nuclear Power Plants, dated October 30, 1979, titled "Discussion of Lessons Learned Short Term Requirements," provided additional clarification of these requirements, and stated "...it is not acceptable to assign a person who is normally the immediate supervisor of the shift supervisor to STA (Shift Technical Advisory duites...".

Contrary to the above, the Chief Operations Engineer, the immediate supervisor of the Senior Watch Supervisor, was assigned to perform STA duties on the 7:00 AM to 3:00 PM, shift of October 17, 1980.

This is a Severity Level III violation. (Supplement I.C.2 of the Interim Enforcement Policy) Civil Penalty - $10,000.

DISCUSSION OF PARAGRAPH IIIA:

A. The duties and responsibilities of the Shift Technical Advisor (STA) are described in Consolidated Edison's Operations Administrative Directive (OAD-9) and in position guides and job descriptions. Among the duties and responsibilities of this position are acting as a technical advisor to the Senior Watch Supervisor in assessing an accident, and evaluating day-to-day plant operations from a safety point of view. To accomplish this, an STA is required to be on shift at all times (within 10 minutes reporting time to the Central Control Room). The Shift Technical Advisor is called to the Control Room (if he is not already there) when problems are identified requiring his expertise and he does participate directly in the assessment of plant conditions. At no time is the STA responsible for manipulation of reactor controls. All of the above is consistent with NRC requirements for the STA.

The Consolidated Edison position guide for the STA explains that assignment to activities not involving engineering review or evaluation is intended and desired during major plant outages. This participation in routine operational and maintenance activities while the plant is shutdown is the best way to continuously expand the knowledge and operational experience of the STAs and familiarize them with plant design, layout and equipment.

Both the latest NRC Standard Technical Specifications (dated August 12, 1980) and the current NRC clarification document regarding TMI Lessons Learned implementation (NUREG-0737, dated October 31, 1980) explicitly state that the Shift Technical Advisor is not required to be on-shift when a plant is in the cold shutdown or refueling condition. Therefore, the use of the STAs for other than their specified functions during cold shutdown outages is consistent with the intended use of the STA, as specified in the NRC's own documents.

The adequacy of the performance and activities of the Shift Technical Advisors during the time period from 11 PM on October 16, 1980 until 3 PM on October 20, 1980 has been reviewed in the course of the Company's own investigation. Although the STA did not recognize the full range of possible effects of the water on reactor systems, the duties and responsibilities as particularly set forth in OAD-9 and the position guides and job descriptions were fulfilled. This conclusion is based on the following:

1. There was a Shift Technical Advisor on duty at all times, within 10 minutes reporting time to the Central Control Room.

2. The shift technical advisors monitored Central Control room activities, e.g., when the problem with nuclear instrumentation channel N42 was observed, the STA independently assessed the situation and discussed this matter with the Senior Watch Supervisor. The STA evaluated the situation as being understood, controlled and addressed properly.

3. The STAs made entries into containment and were aware of the accumulation of water on the containment floor. They had independently evaluated the plant to be in a stable condition, and had evaluated plant operations from a safety point of view. They were aware of the discussions and the actions to be taken regarding the problem of removing the water from the containment building and of correcting the leaks. They were aware of plant conditions and future plans.

The adequacy of our STA directives and training program has been reviewed and we consider them to be consistent with NRC philosophy. These directives include the training program (required by the NRC to be completed by the end of 1980) which was still in progress on October 17, 1980. (In fact, the accident assessment management portion of the STA training program was scheduled for

the week of November 10, 1980.) As required by NRC's NUREG-0737 and previous TMI documents, training of the STAs is to be completed by January 1, 1981. It is felt that this specialized training program coupled with on-the-job experience will result in highly qualified personnel capable of performing the shift technical advisor duties as outlined by the NRC and our own directives.

The STA program at Indian Point 2 therefore does satisfy all NRC requirements and directives, and the actions of the STAs during this event were consistent with their specified functions and the extent to which they had completed their ongoing training program.

We accordingly do not concur in the finding of a violation in paragraph III.A of the Notice of Violation, and deny that such a violation occurred. This item of noncompliance would not in any event be a Severity Level III violation for the reasons set forth at pp. 7-10 above. There was no violation of NRC requirements. NCR Staff's allegation represents an unwarranted attack on the STA training program and the performance of the STAs.

Paragraph III.A of the Notice is incompatible with the findings contained in a report prepared after an NRC Region I inspection of Con Edison's "utility management and technical competence", dated September 2, 1980. With respect to the STA position, the findings of the NRC team were that:

> The STA Training Program is progressing with a completion date commensurate with the dates established in 4a(1) above. Two STA's in training were interviewed to obtained their views regarding the training program, and their perception at this period in time, regarding the the STA position. The STA's interviewed responded positively regarding the training program as conducted thus far, and no specific problems were identified regarding the position and role of the STA.
>
> No items of noncompliance or deviations were identified.
>
> Inspection 50-247/80-11 Section 4(b) at p.3

If contrary to the findings of the NRC staff in September 1980, there were any deficiencies in the STA program, in October, they could not properly be considered Severity Level III, since there were no "violations which involve actual or high potential impact on the public" (45 FR at 66755)

DISCUSSION OF PARAGRAPH III.B:

The NRC's requirement for a Shift Technical Advisor program is intended to be an interim measure until the qualifications of operations personnel such as the shift supervisors and reactor operators have been upgraded. At that time, the operations personnel would have both the desired technical capabilities and the responsibility for manipulation of controls and command of operations. The Chief Operations Engineer who performed the STA duties on the 7 AM to 3 PM shift on October 17, 1980 possesses all of the ultimate qualifications envisioned by the NRC for operations personnel prior to phasing out the STA program. He is an SRO license holder, and a degreed engineer with years of operations experience who is fully familiar with the plant design and layout. He is a trained Emergency Director, who [illegible] a number of years was responsible for the simulator training program at Indian Point. This individual's qualifications, in essence, fully meet the NRC's long-term vision for operations personnel.

Nonetheless, Consolidated Edison admits that this person, in fulfilling the STA function from 7 AM to 3 PM on October 17, 1980, was not supposed to be doing so according to an NRC letter dated October 30, 1979. We thus acknowledge that a violation is stated by paragraph III.B of the Notice of Violation. We do not agree that this is a Severity Level III

violation, there being no safety significance, much less "actual or high potential impact on the public," resulting from this individual having served as STA on October 17, see pp. 7-10 above.

ACTION TO PREVENT RECURRENCE:

The interim practice of the Chief Operations Engineer being designated as STA while the STAs were in initial training during 1980 has been discontinued. Although consistent with present practice, the procedure permitting use of STAs for duties other than their specified functions will be immediately clarified to permit such activity only when the plant is at cold shutdown.

The role of the STA will be strengthened by having the STAs report to the Technical Engineering Director instead of the Chief Operations Engineer. This will be accomplished prior to return to service.

Since the establishment of the STA function in 1979, there has been continuing discussion and evolution of the specific role the STA is to play in plant operation. The definition of what constitutes a "problem" requiring STA evaluation, and the STA's ability to recognize the potential significance of all types of events within his purview, are both improving with training and experience. The circumstances leading up to and surrounding this event will be used in future lesson plans for ongoing training/retraining of operations personnel, including the STAs. The revision to the training program to accomplish the above will be completed prior to return to service.

OFFICE OF INSPECTION AND ENFORCEMENT STATEMENT OF ALLEGED VIOLATION

IV. Technical Specification 6.8.1 requires that: "Written procedures shall be established, implemented and maintained..." Procedure E-12, "Nuclear Instrument Malfunction", Rev. 3 dated 7/5/78, step C-4.1.3 requires as "Immediate Operator Action", if one channel fails, that C-5.5 of Procedure E-12 subsequently requires that all the nuclear bistables associated with the defective channel be tripped by removing the control power fuses.

Contrary to the above: On October 17, 1980, the licensee removed the control power fuses associated with the defective channel N42, with reactor power level at about 90%. This resulted in an automatic runback to less than 75% reactor power.

This is a Severity Level V violation (Supplement I.E of the Interim Enforcement Policy).

DISCUSSION:

The procedure for responding to malfunctioning power range detectors is based primarily on sudden failures calling for immediate operator action. The failure in question was characterized as a gradual drift in one of the four power range detection signals over a period of hours. The operator had calculations performed and tests conducted which proved that the signal from the detector in question was false, and that the reactor power distribution was normal.

Recognizing that a dropped control rod condition did not exist and that there was proper reactor power distribution, the operator proceeded to the subsequent action section of the procedure. One step in this section calls for removing control power fuses. The operator failed to recognize that removal of these fuses would result in an automatic reduction in turbine power to 70%. When the fuses were removed, the turbine power "ran back" as designed.

Consolidated Edison accordingly acknowledges that a violation is stated in paragraph IV of the Notice of Violation. We do not agree, however, that the violation is properly considered as a Severity Level V under the Interim Enforcement Policy. This severity level applies to violations with "other than [i.e., more than] minor safety significance," whereas the Notice of Violation does not identify any safety significance associated with this matter. In fact, the turbine runback to 70% power did not even result in a trip of the plant.

ACTION TO PREVENT RECURRENCE:

A step has been added to the subsequent action section of the procedure cautioning the operator to verify that the plant is at, or below, 70% turbine load before the control power fuses are removed. An additional precautionary note has also been added to the immediate automatic action section alerting the operator to the possibility of instrument drift in the high or low direction.

These procedural changes were accomplished by October 31, 1980. Retraining with the new procedure, which is ongoing, will emphasize the need to strictly adhere to all procedural requirements, and will be completed prior to reactor startup.

OFFICE OF INSPECTION AND ENFORCEMENT
STATEMENT OF ALLEGED DEVIATION

Contrary to standard industry practice and the manufacturer's Technical manual, "Goulds Installation, Operation, and Maintenance Instructions for Vertical Sump Pumps, Models 3171, 3172, 3173, 3174" the containment sump pump float rods were not attached or guided at the bottom from October 17, 1980 through October 20, 1980. This contributed to sump pump inoperability during the containment flooding incident. Also, contrary to guidance on page 9 of the manufacturer's Technical Manual, the pumps were not prevented from running against a shutoff head on September 14, 1980 and September 15, 1980 and at various times from October 17, 1980 to October 19, 1980 when the pump discharge valves were shut and power to the pumps was not secured.

In reply, please comment on this item, including a description of all actions that have been or will be taken to correct the item and prevent recurrence and the date when these actions have been or will be completed.

DISCUSSION:

The pumps installed during original plant construction were provided with a single upper guide rod and prior to this incident have operated satisfactorily since initial plant operation.

The concern in operating centrifugal pumps for an extended period of time against a closed discharge valve damage they might incur from overheating and/or pumped fluid vaporization. Since the containment sump pumps are completely submerged, however, there is adequate cooling. The pumps have never exhibited problems operating against a closed discharge valve. Examination of the removed pumps showed no significant wear to pumps or motors. In fact, neither of the containment sump pumps failed in the course of the flooding incident; they did not pump because they did not receive electric power.

We have been informed by the pump manufacturer that they normally recommend a minimum flow to assure prolonged bearing life. They advised that operation at shutoff conditions will not cause a catastrophic type failure with the model 3171 pump and that a result of this type of operation could be increased steady bearing wear at the adapter caused by higher radial thrust loading at the

impeller. We examined the pumps following this event and no evidence of such wear was observed.

Consolidated Edison accordingly does not concur that a deviation is set forth in Appendix B to the Notice of Violation.

ACTIONS TO PREVENT RECURRENCE:

To eliminate any potential for excessive bearing wear while operating the Containment Sump Pumps against shutoff head conditons, a recirculation line with an orifice will be provided from each pump discharge back to the containment sump. This will assure minimum flow through each pump for all operating conditions.

Improved level control devices for start/stop of sump pumps will be installed to replace the existing integral float control assemblies for each containment sump pump to eliminate guide rod binding. Additionally, containment sump pump control and indication will be be installed in the Control Room to provide additional system status information to the operator and the capability for remote operation of each pump as a backup to normal automatic control.

All of the above will be completed prior to return to service.

SUMMARY

Alleged Violation I.A is admitted under the interpretation of "expected condition" employed in the Notice of Violation, which Consolidated Edison submits is vague and indefinite, and disregards the "serious event" context of this reporting requirement as it had been contained in previous NRC guidance.

Alleged Violation II.A is admitted, although the Station Nuclear Safety Committee was unaware of the wetting of the reactor vessel and associated components on October 20, 1980.

Alleged Violation II.F is admitted. The inadvertent misidentification of two very similar materials was an isolated circumstance occuring in 1975, which did not contribute to the October leakage event.

Alleged Violation III.B is admitted, although the individual fulfilling the Shift Technical Advisor role during the eight hour period charged was fully qualified to perform the STA function. This violation was also unrelated to the leakage event.

Alleged Violation IV, the inadvertent causing of a turbine runback with no other consequences, is admitted.

Consolidated Edison denies all of the remaining violations alleged in the Notice of Violation. The Company also denies

that either the October 1980 leakage event or its operation of Indian Point Unit 2 during the time referred to in the Notice brought about any actual or high potential safety impact on the public. We accordingly contend that the violation Severity Levels set forth in the Notice are erroneous, and do not conform to the criteria set forth in the Interim Enforcement Policy (45 FR 66754). Moreover, the number of violations for which Consolidated Edison is cited is not in accordance with the Interim Enforcement Policy, particularly in regard to the progression of escalated enforcement actions (45 FR at 66758).

For these reasons and the reasons set forth in the separate discussions of the various alleged violations above, Consolidated Edison requests remission or mitigation of any penalty proposed in accordance with the Interim Enforcement Policy. Please see our concurrent response supplied pursuant to 10 CFR 2.205(b).

John D. O'Toole

John D. O'Toole
Assistant Vice President
Consolidated Edison Company
of New York, Inc.

Dated: New York, New York
January 5, 1981

STATE OF NEW YORK §
§
COUNTY OF NEW YORK §

JOHN D. O'TOOLE, being duly sworn, deposes and says: That he is an Assistant Vice President of CONSOLIDATED EDISON COMPANY OF NEW YORK, INC., Licensee of Indian Point Unit 2 herein; that the foregoing Statement in Reply to Notice of Violation dated January 5, 1981 has been prepared under his supervision and direction; that he knows the contents thereof; and that to the best of his knowledge and belief said reply and the facts contained therein are true and correct.

DATED: New York, New York
January 5, 1981

John D. O'Toole

John D. O'Toole

Subscribed and sworn to before
me this 5th day of January, 1981

Susan D. Barotsky

Notary Public

Suffolk

Accession No.:
8008220270
IEB 80-24

UNITED STATES
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

November 21, 1980

IE Bulletin No. 80-24: PREVENTION OF DAMAGE DUE TO WATER LEAKAGE INSIDE CONTAINMENT (OCTOBER 17, 1980 INDIAN POINT 2 EVENT)

Description of Circumstances:

On October 24, 1980 IE Information Notice No. 80-37 described an event that occurred at the Indian Point Unit 2 (IP-2) facility. On October 17, 1980, upon containment entry for repair to a nuclear instrument, it was discovered that several inches of water had accumulated on the containment floor without the operators' knowledge. This accumulation was later determined to have amounted to over 100,000 gallons which flooded the reactor vessel pit and wetted the lower nine feet of the reactor vessel while the reactor was at operating temperature.

The flooded condition resulted from the following combination of conditions: (1) There were significant multiple service water leaks from piping and fan coolers onto the containment floor. This system had a history of leakage; (2) Both containment sump pumps were inoperable, one due to blown fuses and the other due to binding of its float switch; (3) The significance of two containment sump level indicating lights which indicated that the water level was continuously above the pump-down level was not recognized by the operators; (4) There was no high water level alarm and the range of sump level indicating lights failed to indicate the overflowing sump level; (5) The moisture level indicators for the containment atmosphere did not indicate high moisture levels, apparently due to an error in calibration and/or ranging which made them insensitive to the moisture levels resulting from relatively small cold water leaks; (6) The hold-up tanks which ultimately receive water pumped from the containment sump also received water from other sources (Unit 1 process water, lab drain water, etc). These other water sources masked the effect of cessation of water flows from the Unit 2 sump; (7) The fan cooler condensate wier level measuring instruments were not properly calibrated; (8) There was no water level instrumentation in the reactor vessel pit and the pumps were ineffective since they discharge to the containment floor for ultimate removal by the containment sump pumps.

This Bulletin is issued to enable the NRC staff to formulate requirements for long term generic corrective actions which will be the subject(s) of future NRC actions. The bulletin requires short term actions which will preclude IP-2 type events at other plants in the interim before the longer term generic actions are accomplished.

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Actions to be Taken by Licensees:

1. Provide a summary description of all open* cooling water systems present inside containment. Your description of the cooling water systems must include: (a) Mode of operation during routine reactor operation and in response to a LOCA; (b) Source of water and typical chemical content of water; (c) Materials used in piping and coolers; (d) Experience with system leakage; (e) History and type of repairs to coolers and piping systems (i.e., replacement, weld, braze, etc.); (f) Provisions for isolating portions of the system inside containment in the event of leakage including vulnerability of those isolation provisions to single failure; (g) Provisions for testing isolation valves in accordance with Appendix J to 10 CFR 50 (h) Instrumentation (pressure, dew point, flow, radiation detection, etc.) and procedures in place to detect leakage; and (i) Provisions to detect radioactive contamination in service water discharge from containment.

2. For plants with open cooling water systems inside containment take the follow actions:

   a. Verify existence or provide redundant means of detecting and promptly alerting control room operators of a significant accumulation of water in containment (including the reactor vessel pit if present).

   b. Verify existence or provide positive means for control room operators to determine flow from containment sump(s) used to collect and remove water from containment.

   c. Verify or establish at least monthly surveillance procedures, with appropriate operating limitations, to assure plant operators have at least two methods of determining water level in each location where water may accumulate. The surveillance procedures shall assure that at least one method to remove water from each such location is available during power operation. In the event either the detection or removal systems become inoperable it is recommended that continued power operation be limited to seven days and added surveillance measures be instituted.

   d. Review leakage detection systems and procedures and provide or verify ability to promptly detect water leakage in containment, and to isolate the leaking components or system. Periodic containment entry to inspect

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* An Open system utilizes an indefinite volume, such as a river, so that leakage from the system could not be detected by inventory decrease. In addition, a direct radioactive pathway might exist to outside containment in the event of a LOCA simultaneous with a system leak inside containment. A closed system utilizes a fixed, monitored volume such that leakage from the system could be detected from inventory decrease and a second boundary exists to prevent loss of containment integrity as a result of a system leak inside containment.

for leakage should be considered.

e. Beginning within 10 days of the date of this bulletin, whenever the reactor is operating and until the measures described in (a) through (d) above are implemented, conduct interim surveillance measures. The measures shall include where practical (considering containment atmosphere and ALARA considerations) a periodic containment inspection or remote visual surveillance to check for water leakage. If containment entry is impractical during operation, perform a containment inspection for water leakage at the first plant shutdown for any reason subsequent to receipt of this bulletin.

f. Establish procedures to notify the NRC of any service water system leaks within containment via a special licensee event report (24 hours with written report in 14 days) as a degradation of a containment boundary.

3. For plants with closed cooling water systems inside containment provide a summary of experiences with cooling water system leakage into containment.

4. Provide a written report, signed under oath or affirmation, under the provisions of Section 182a of the Atomic Energy Act of 1954, in response to the above items within 45 days of the date of this bulletin. Include in your report where applicable, your schedule for completing the actions in response to items 2 (a) through (d). Your response should be sent to the Director of the appropriate Regional Office with a copy forwarded to the Director, NRC, Office of Inspection and Enforcement, Washington, D.C. 20555.

If you desire additional information regarding this matter please contact the appropriate IE Regional Office.

Approved by GAO, B180225 (R0072); clearance expires November 30, 1980. Approval was given under a blanket clearance specifically for identified generic problems.