William J. Cahill, Jr.
Vice President

Consolidated Edison Company of New York, Inc.
4 Irving Place, New York, N Y 10003
Telephone (212) 460-3819

October 9, 1979

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

Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Denton:

This letter responds to your September 17, 1979 letter on the subject of potential interaction between non-safety grade systems and safety grade systems. This potential problem was further addressed in the IE Information Notice 79-22, issued September 14, 1979.

In conjunction with Westinghouse, we have reviewed the specific non-safety grade systems listed in IE Information Notice 79-22, for potential interactions that could constitute a substantial safety hazard. We have not been able to identify such an interaction. The basic conclusion of the FSAR, that these events do not constitute an undue risk to the health and safety of the public, remains unchanged. The results of the plant specific evaluation justify continued operation of Indian Point Unit No. 2.

We have completed plant specific evaluations, for Indian Point Unit No. 2, of the four control system interaction scenarios postulated by Westinghouse. For the first 3 postulated scenarios, we have found the existing control system equipment qualified and, therefore, these events are not applicable to Indian Point Unit No. 2. For the fourth Westinghouse postulated interaction scenario, Westinghouse has completed a typical realistic analysis the results of which are within the limits established for the FSAR safety analyses (DNBR greater than 1.3). Westinghouse believes that this analysis will be bounding for Indian Point Unit No. 2. Attachment 1 reviews the details of our evaluations.

As a result of the Three Mile Island accident, there are a significant number of industry, governmental and regulatory investigations under way examining the licensing bases and the operating procedures of nuclear generating facilities. These investigations are already identifying areas where studies may result in the consideration of new or revised events as part of the bases for assuring the continued safety of nuclear plants. NUREG-0578 outlines several such events and suggests remedies.

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NUREG-0578 requirements for analyses of potential safety problems envision the kinds of scenarios identified by Westinghouse and made the subject of IE Information Notice 79-22. Section 3.2, page 17 states in part,

> "...The NRC requirements for non-safety systems are generally limited to assuring that they do not adversely affect the operation of safety systems..."

Further, on page A-25 of NUREG-0578,

> "Consequential failures shall also be considered..."

We, therefore, believe that the scope of the action required by IE Information Notice 79-22 is consistent with the requirements of NUREG-0578 and should therefore be integrated with the planned response sequence for compliance with the NUREG.

This information is being submitted pursuant to 10CFR50.54(f) and forty (40) copies of this submittal are being provided.

Should you or your staff have any additional questions, please contact us.

Very truly yours,

William J. Cahill, Jr.
Vice President

attach.

Subcribed and sworn to
before me this 9th day
of October, 1979.

Angela Roberti
Notary Public
ANGELA ROBERTI
Notary Public, State of New York
No. 41-8593813
Qualified in Queens County
Commission Expires March 30, 1980

ATTACHMENT 1

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION ON THE ENVIRONMENTAL INTERACTION ISSUE

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
Facility Operating License No. DPR-26
October, 1979

ATTACHMENT 1

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION ON THE ENVIRONMENTAL INTERACTION ISSUE

Summary

The information contained in this attachment justifies continued operation of Indian Point Unit No. 2 on the basis of the qualification of the Control System equipment to operate in the postulated accident environment, the improbability of the postulated scenarios as they apply to Indian Point Unit No. 2, and the acceptability of the consequences.

Scope

On 9/18/79 Westinghouse presented to the Staff a summary of the investigation that had been conducted which led to the identification of four (4) potential interaction scenarios where the effect on control systems of adverse environments resulting from high energy line breaks, could lead to consequences more limiting than the results presented in the Safety Analysis Report. Table 1 summarizes the scope of the Westinghouse investigation.

The seven (7) control systems selected for the investigation by Westinghouse include all control systems directly addressed in the current Westinghouse functional requirements. The seven (7) accidents considered encompass all postulated High Energy Line Break (HELB) environments, including all break locations and a range of break sizes. Of the forty-nine (49) combinations of control systems and accident environments investigated, fifteen (15) interaction scenarios, denoted by an X in Table 1, were identified which resulted in consequences more severe than reported in the Safety Analysis Reports. Westinghouse confirms that the fifteen interactions identified are bounded by consideration of the four (4) interactions discussed herein. The following section discusses the applicability of these postulated scenarios with respect to Indian Point Unit No. 2, and concludes that they are not applicable.

Applicability of Identified Scenarios

This section establishes that the four scenarios postulated by Westinghouse are not applicable to Indian Point Unit No. 2.

I. STEAM GENERATOR PORV CONTROL SYSTEM

a) Summary of Postulated Westinghouse Scenario

Following a feedline rupture outside containment in the auxiliary building, the steam generator PORV's are assumed to exhibit a consequential failure due to an adverse environment. Failure of the PORV's in the open position results in the depressurization of multiple steam generators which are the source of steam supply for the turbine driven auxiliary feedwater pump. Eventually, the turbine driven auxiliary feedwater pump will not be capable of delivering auxiliary feedwater to the intact steam generators. Depending upon auxiliary system design, a potential exists that no auxiliary feedwater will be injected into the intact steam generators until the operator takes corrective action to isolate the auxiliary flow spilling out the rupture.

b) Plant Specific Evaluation

The postulated break is outside containment between the check valve (elevation 43') and the containment penetration. This is approximately 30 feet of Class I seismic pipe. Con Edison has previously addressed breaks in this location. A report, "Analysis of High Energy Lines" was submitted to the NRC for Indian Point Unit No. 2 by a letter from Leonard Trosten (LeBoeuf, Lamb, Leiby & MacRae) to Angelo Giambusso (NRC), dated April 9, 1973. That analysis included drawings of the piping addresed in our current scenario and analyzed the same break in Section 4.3, "Shield Wall Area at Elevation 43'".

Con Edison has identified steam generator PORV control system components that will experience a severe environment as a result of a break in one of these pipes. There are control system panels on either side of the pipes in this area, and mounted on the panels are Fisher Type 546 Electro Pneumatic Transducers (I/P Converters) and Fisher Type 67 FR Pressure Regulators which control the Steam Generator PORV's. We have contacted Fisher and obtained a June 12, 1973 Fisher Laboratory Report which describes tests that included both of these Fisher devices. The testing exposed the equipment to a temperature of 320°F and 75.3 PSIG for a period of one hour, followed by a twelve hour period at 288°F.

Con Edison has calculated a temperature of 213°F for impingement at the break in the Main Feedwater Piping. Therefore we conclude that the Fisher equipment is more than adequately qualified for the environment that it might experience for this scenario since it successfully withstood the test environment.

This postulated scenario, therefore, does not apply to Indian Point Unit No. 2.

II. MAIN FEEDWATER CONTROL SYSTEM

a) Summary of Postulated Westinghouse Scenario

Following a small feedline rupture, the main feedwater control system malfunctions in such a manner that the liquid mass in the intact steam generators is less than for the worst case presented in Safety Analysis Reports. The reduced secondary liquid mass at time of automatic reactor trip results in a more severe reactor coolant system heatup following reactor trip.

b) Plant Specific Evaluation

The postulated break is a small rupture in the main feedwater lines or a break in the auxiliary feedwater lines. This is the same location in the auxiliary feedwater building as the first scenario. The "Analysis of High Energy Lines" report previously referred to also applies. The qualified equipment identified in the previous scenario would still be qualified for this scenario.

Con Edison has identified main feedwater control system equipment that could experience a severe environment as a result of a break in one of these pipes. These are the ASCO solenoid valves that control the Main Feedwater Regulator Valves. These ASCO solenoid valves are qualified for 145°F for long term continuous energized service. They are also qualified to 286°F for 1/2 hour per the ASCO letter attached to Westinghouse letter NS-CE-755, dated August 15, 1975 and submitted to the NRC as part of the Environmental Qualification Program. Con Edison has contacted ASCO representatives. They confirm that these ASCO solenoid valves are qualified for 200°F for 8 hours.

Since these ASCO solenoid valves are more than 50 feet away from the area of the postulated pipe break and at a lower elevation, there is no possibility of direct impingement. Con Edison has calculated an auxiliary feedwater building ambient temperature from the worst feedline break of 160°F.

Therefore we conclude that the ASCO solenoid valves are qualified for the severe environment that they might experience for the postulated breaks in this scenario.

This postulated scenario, therefore, does not apply to Indian Point Unit No. 2.

III. PRESSURIZER PORV CONTROL SYSTEM

a) Summary of Postulated Westinghouse Scenario

Following a feedline rupture inside containment, the pressurizer PORV control system malfunctions in such a manner that the power operated relief valves fail in the open position. Thus in addition to a feedline rupture between the steam generator nozzle and the containment penetration, a breach of the reactor coolant system boundary has occurred in the pressurizer vapor space.

b) Plant Specific Evaluation

The postulated break is in containment, in the main feedwater lines. Con Edison has already identified pressure transmitters and cable as equipment subject to a severe environment and they are environmentally qualified for in containment service. Reference the Indian Point Unit No. 2 FSAR, Answer to Question 7.8, Supplement 8, August 1970. For the pressure transmitters, reference WCAP 7410-L, Volume I, December 1970; and WCAP 7354-L, July 1969. For the cables, reference WCAP 7410-L, Volume II, December 1970.

For a feedline break Con Edison has calculated a 213°F direct impingement temperature at the break, as discussed in the previous scenarios.

In addition the Pressurizer pressure transmitters are located in Rack 19, between the crane wall and the shield wall at elevation 68. The feedwater lines pass thru this crane wall area rising to a higher elevation approximately 90° away on the containment circumference. Therefore there is no possibility of direct impingement.

Therefore the transmitters and cable are qualified for the environment produced by this scenario since they have been qualified for in-containment service at 286°F and 60 psig.

The only other control system equipment that could be exposed to a severe environment due to this postulated break is the ASCO solenoid valves that control the Pressurizer PORVs. ASCO NP-1 solenoids environmentally qualified for in containment use were installed during the summer 1979 refueling/maintenance outage as part of our response to I.E. Bulletin 79-01A.

All control equipment in containment associated with the Pressurizer PORVs that might be subject to a severe environment is qualified for that environment.

This postulated scenario, therefore, does not apply to Indian Point Unit No. 2.

IV. ROD CONTROL SYSTEM

a) Summary of Postulated Westinghouse Scenario

Following an intermediate steamline rupture inside containment, the rod control system if operated on automatic mode, would exhibit a consequential failure due to an adverse environment which causes the control rods to begin stepping out prior to receipt of a reactor trip signal on overpower delta-T. This scenario results in a lower DNB ratio than presently presented in Safety Analysis reports.

b) Plant Specific Evaluation

The conservative assumptions already contained in the Design Basis Event analysis reported in the Safety Analysis Report and the additional conservative assumptions to be made to derive the interaction scenario postulated by Westinghouse, a significantly less probable subset of the Design Basis Events, must occur for the postulated Westinghouse scenario to be applicable for Indian Point Unit No. 2. At this time no determination

has been made as to (1) whether the nuclear instrumentation system equipment should be adversely affected (2) what the failure mode would be and (3) whether the failure would take place prior to reactor trip (i.e., within 2 minutes).

Westinghouse has informed us that they have performed a typical realistic analysis of this postulated scenario which shows that DNBR is greater than 1.3. Westinghouse believes that this typical analysis bounds Indian Point Unit No. 2.

Additionally it should be noted that the plant rarely operates in automatic rod control mode. This significantly reduces the probability that this postulated scenario will occur.

Also, Westinghouse has advised us that they will be conducting an environmental qualification program for the ex-core detectors in early 1980. We will review the results of that program and make appropriate changes.

Considering the improbability of this scenario and the acceptability of the consequences, continued operation of Indian Point Unit No. 2 is justified.

Consequences of Postulated Interaction

We have evaluated for Indian Point Unit No. 2 the four interaction scenarios postulated by Westinghouse and have not been able to identify any interaction which could constitue a substantial safety hazard. The basic conclusion of the FSAR, that these events do not constitute an undue risk to the health and safety of the public, remains unchanged and continued operation of Unit 2 is justified.

Due to the implementation of the design of the electrical separation requirements between control and protection systems specified in IEEE-279, the only interaction mechanisms identified in the above scenarios result from conservatively assuming an adverse environment at the location of the control systems and the consequential equipment failure in the worst directio As a consequence, it can be anticipated that any interaction scenarios yet to be identified, in as yet unreviewed control systems, will be no more probable than the particular scenarios described by Westinghouse to date.

Future Investigations

With regard to future investigation into as yet unreviewed areas where similar interaction mechanisms may be identified, Con Edison will participate in a future on-going program of investigation. This commitement is consistent with the long-term recommendation by the Task Force in NUREG-0578 concerning the furure development of General Safety Criteria.

Control System / Accident	Reactor Control	Pressurizer		Feedwater Control	Steam Generator Pressure Control	Steam Dump System	Turbine Control
		Pressure Control	Level Control				
Small Steamline Rupture	X	X			X		
Large Steamline Rupture		X			X		
Small Feedline Rupture	X	X		X	X		
Large Feedline Rupture	X	X			X		
Small LOCA	X	X		X			
Large LOCA							
Rod Ejection							

TABLE 1

PROTECTION SYSTEM-CONTROL SYSTEM POTENTIAL ENVIRONMENTAL INTERACTION

X - POTENTIAL INTERACTION IDENTIFIED THAT COULD DEGRADE ACCIDENT ANALYSIS

☐ - NO SUCH INTERACTION MECHANISM IDENTIFIED