April 26, 1995

Mr. Stephen E. Quinn
Vice President, Nuclear Power
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenue
Buchanan, NY 10511

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION (RAI), INDIVIDUAL PLANT EXAMINATION (IPE) SUBMITTAL, INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 (TAC NO. M74422)

Dear Mr. Quinn:

By letter dated August 12, 1992, the Consolidated Edison Company of New York, Inc. (Con Edison) submitted the IPE for the Indian Point 2 (IP2) plant in response to NRC Generic Letter 88-20, Supplement 1. The NRC has now completed its initial review of the IPE and has identified additional information required. Con Edison is requested to respond to the enclosed RAI to enable us to complete the review. If your response will take longer than 60 days, please advise us of your schedule for submittal. If you have any questions regarding this RAI, please contact me at 301-415-1412.

This requirement affects one respondent and, therefore, is not subject to Office of Management and Budget review under P.L. 96-511.

Sincerely,
Original signed by D. McDonald for:
Francis J. Williams, Jr., Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-247

Enclosure: Request for Additional Information

cc w/encl: see next page

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REQUEST FOR ADDITIONAL INFORMATION ON THE

INDIVIDUAL PLANT EXAMINATION (IPE) SUBMITTAL

INDIAN POINT 2 FRONT END QUESTIONS

1. Please provide a discussion of the process used to arrive at a list of plant specific initiating events, over and above those used in the Indian Point Probabalistic Safety Study (IPPSS). Provide the reasons for including or excluding initiators; if based on frequency, provide the criteria used. Please include in your discussion the following:

   a) Losses of electrical buses: 480V ac, 6.9kV ac, 118V ac and 125V dc. Also provide the basis on which common cause failure for 2 or more buses was eliminated.

   b) Loss of heating, ventilation, and air conditioning (HVAC). This is an important initiator at similar plants. However, there is no discussion of these events at Indian Point 2. Please provide a discussion of your investigation into impact of loss of HVAC in rooms containing safety-related equipment, including rooms with pumps, electrical equipment (ac and dc) and the control room. Your discussion should include the following: systems in the areas considered; basis for elimination, describing the method of assessment, including calculations and tests; credited operator actions, alarms, procedures and staged equipment.

   c) Loss of offsite power (LOOP). Please discuss whether a distinction is made between a LOOP at the unit, and an event that also affects the other unit on site (Indian Point 3). Identify which is considered for the IP2 IPE. The concern is that for a site-wide LOOP, the other unit may need to use some of the systems credited in the IPE. What is the frequency of the site-wide LOOP and what would be the impact of considering such an event on the core damage frequency?

2. The following questions pertain to the common cause data section:

   a) The common cause data list is referred to in the submittal (Table 3.3-5) but is missing (Table 3.3-5 in the submittal is for preaccident human error probablities (HEPs). Please provide this list.

   b) It is not clear from the submittal that the licensee's list of common cause events includes important components that could be important in identifying potential vulnerabilities. For example, the following equipment categories are not included:

Enclosure

circuit breakers,
electrical switch gear and buses,
batteries,
inverters, and
check valves (fail to open).

Provide the technical basis for the omission of these components from the common cause analysis. Explain how it was assured that vulnerabilities were not missed as a result of the omission of these common cause failures.

3. This question concerns the reactor coolant pump (RCP) seal loss-of-coolant accident (LOCA) model. Please discuss the following:

   a) Please identify and provide a description of the seal LOCA model used in your IPE. Provide specifics about the seal LOCA model used (e.g. seal leakage rates and probability of seal failure vs. time to seal failure).

   b) Can loss of either seal injection or thermal barrier cooling lead to vibration problems, which could induce a seal LOCA? If yes, how was this considered in the seal LOCA modeling?

4. The IPE submittal is not clear as to whether one residual heat removal (RHR) or one low-pressure recirculation (LPR) pump is sufficient to provide recirculation decay heat removal. In certain cases (e.g., in large LOCAs when fan coolers are unavailable), these same pumps will also be called on to supply the flow to the containment spray headers. Please discuss the following:

   a) In the recirculation phase, is one RHR or one LPR pump sufficient to provide enough flow for both core coolant recirculation and the spray headers? Please discuss how this was modeled in the analysis, including operator actions necessary to distribute the flow between core cooling and containment spray recirculation.

   b) How is long term fouling of RHR heat exchangers, as experienced at many plants (evidenced by testing to meet the requirements of Generic Letter 89-13) accounted for in the model? If not accounted for, please discuss the expected impact this would have on RHR or recirculation pump temperature, net positive suction head (NPSH), and pump operation, especially the ability to provide sufficient flow for both core coolant and spray recirculation referred to in (a) above.

5. This question addresses the success criteria which were used for various initiators. Since the NRC reporting guidelines include confirmation that the IPE represents the "as built as operated plant," please verify that the Modular Accident Analysis Program (MAAP) calculations and the

Westinghouse Commercial Atomic Power (WCAP) references address the current as built and as operated plant and describe how this was ascertained.

6. Some IPE initiating event frequencies are much smaller than the ones used in the original IPPSS, such that even the 95th percentile is smaller than the mean IPPSS value. For instance, turbine trip frequency has been reduced from 7.3/yr to 1.3/yr, loss of main feedwater has been reduced from 6.7/yr to 1.3/yr and main steam isolation valve (MSIV) closure has been reduced from 1.3/yr to 0.118/yr. Please explain why there is such a significant reduction in these initiating event frequencies.

7. The submittal identifies a number of improvements which were going to be addressed. However there is no discussion about the dates the improvements will be implemented. Please identify the status of these modifications, and, if available, the reduction in the estimated core damage frequency due to these modifications.

8. It is stated that the criteria used to determine whether core damage occurred was either 2200 °F peak cladding temperature or core uncovery. Please clarify which criteria are used for which initiators and why. Discuss whether or not the difference in the criteria makes a difference in core damage frequency.

9. It is not clear how the pressurizer power operated relief valves (PORVs) and the block valves are modeled in the IPE. Please discuss:

   a) What fraction of time is one or both block valves closed?

   b) How is this modeled (specifically when feed and bleed is called for or in anticipated transient without scram (ATWS) sequences when pressure relief is needed)?

   c) Please discuss the operator actions required to open the block valves and the PORVs when needed.

10. In the diagram of the electrical system (Figure labeled 8.2-3 in Chapter 3), there seems to be a capability for switching between the buses and the sources of power (e.g. emergency buses can be cross connected within the unit). Was credit taken for these within-unit bus cross-ties? If so, please discuss, including possible downside effects.

11. This question concerns completeness of treatment of any multi-unit effects from the other Indian Point units on site. For instance, on page 3-116, section 3.2.15.2, it is stated that "some of the balance of plant equipment serviced by the non-essential (service water) header may also be serviced from an alternate source in the retired Indian Point Unit No. 1 facility." The use of this equipment is not included in the modeling of this system. On page 3-89, section 3.2.3.2, it is stated

that "the backup supply to the auxiliary feedwater pumps is the City Water Storage Tank which has a 1.5 million gallon capacity and is shared with Indian Point Unit No. 3."

Are there any additional systems which may be shared or cross-connected with other units on site? If there are, please discuss the following:

a) How are these shared systems incorporated into the model?

b) Describe the operator actions that would be required to use these systems.

c) Describe any possible downside effects arising from the multi-unit plant site.

12. Describe how the city water system is modeled in the analysis, including equipment used, dependencies, and operator actions. Please provide the HEPs for the operator actions and the quantification of the HEPs.

13. Initiating event frequencies, for the loss of 6.9 kV buses 2 and 3 and for interfacing-systems loss-of-coolant accident (ISLOCA) are missing from Table 3.3-1, "Initiating Event Frequencies" Please provide these frequencies.

14. It is not clear what is meant by "loss of offsite power," and exactly in what situations it is possible to recover by using the gas turbines. For instance, it is possible that LOOP originates by failure of certain switchgear equipment (transformers, breakers), which would make it impossible to connect the gas turbines to the plant, and therefore, in certain fraction of cases no recovery is possible (other than repair of switchgear equipment or diesel recovery). Please discuss how these different types of failures were factored into your model.

15. Recovery of a single gas turbine will apparently not lead to recovery of all the emergency buses, and therefore, certain safety equipment may not be powered in that case. Please explain how this is accounted for in the power recovery model and in the event tree.

16. It is not clear in the submittal if plant changes due to the Station Blackout rule were credited in the analysis. Please provide the following: (1) identify whether plant changes (e.g., procedures for load shedding, alternate AC power) made in response to the blackout rule were credited in the IPE and what are the specific plant changes that were credited; (2) if available, identify the total impact of these plant changes to the total plant core damage frequency and to the station blackout CDF (i.e., reduction in total plant CDF and station blackout CDF); (3) if available, identify the impact of each individual plant change to the total plant core damage frequency and to the station blackout CDF (i.e., reduction in total plant CDF and station blackout CDF); (4) identify any other changes to the plant that have been

implemented or planned to be implemented that are separate from those in response to the station blackout rule, that reduce the station blackout CDF; (5) identify whether the changes in #4 are implemented or planned; (6) identify whether credit was taken for the changes in #4 in the IPE; and (7) if available, identify the impact of the changes in #4 to the station blackout CDF.

INDIAN POINT 2 HUMAN RELIABILITY QUESTIONS

PREINITIATOR HUMAN ERRORS

17. In Subsection 3.3.3.2, "Pre Accident Human Interactions," of the submittal, it states that "The specific pre-accident human interactions included in the IP2 IPE are listed in Table 3.3-6." The table, entitled "Overview of Pre Accident Operator Actions in the Indian Point 2 IPE," briefly describes 11 different preinitiator operator (or maintenance) action events along with associated HEPs. Each event is associated with one of four specifically identified systems. No calibration events are included in Table 3.3-6. The submittal is not clear whether the impact of a human to cause an accident from disabling a system due to miscalibration of critical instrumentation was considered.

    The submittal does not clearly identify the events or discuss the process that was used to identify and select preinitiator human events involving failure to properly restore to service after test or maintenance and miscalibration of critical instrumentation. The process used to identify and select these types of human events may include the review of operations, maintenance, test and calibration procedures and discussions with appropriate plant personnel on interpretation and implementation of the these procedures.

    Please list and describe in detail the preinitiator human events that were considered in the IPE including the 11 events in Table 3.3-6, specifying if the events included: a) failure to properly restore equipment to service after test or maintenance or b) miscalibration of critical instrumentation.

    In addition, please provide a description of the process that was used to identify these errors with several examples illustrating this process.

18. In Subsection 3.3.3.2 of the submittal, it states: "When two or more tasks are performed, dependence between human errors must be addressed." It further states that Swain's Handbook (NUREG/CR-1278) "... defines five levels of dependence and provides equations for evaluating the conditional probability of failure ..." for low, moderate, high and complete dependence. None of the 11 preaccident human events in Table 3.3-6, "Overview of Pre Accident Operator Actions in the Indian Point 2 IPE," are noted specifically as being dependent events. (Nevertheless, the table does contain two sets of preaccident events related to two or

three emergency diesel generators (EDGs), namely failure to "realign one, two or three EDG control switch(es) following test" and "undiscovered maintenance error on two or three EDGs.")

It is not clear from the submittal how dependencies associated with preinitiator human errors were identified and evaluated. Failure to identify and evaluate different types of preinitiator human events dependencies that could potentially exist can result in failure to recognize vulnerabilities associated with the design, operation, maintenance or surveillance testing of the plant. In addition, whether miscalibration or failure to restore, the process utilized should consider plant conditions, human engineering, performance by same crew at same time, adequacy of training, adequacy of procedures, and interviews with training, operations and various crews.

Please provide a brief discussion on what dependencies were identified, how they were identified, evaluated and treated in the preinitiator human reliability analysis (HRA) such that important accident sequences were not eliminated.

POST-INITIATOR HUMAN ERRORS

19. Subsection 3.3.3.3, "Post Accident Human Interactions," of the submittal is apparently, by title, associated with Table 3.3-7, "Overview of Post Accident Operator Actions in the Indian Point 2 IPE" (even though the table is not referred to in the subsection). The table briefly describes a total of 36 different post-accident operator and maintenance action events along with their associated HEPs, time windows (T_w) and probablistic risk assessment (PRA) codes. Based on the description of the post-accident human events and their codes, the IPE apparently has used other human events besides those provided in Table 3.3-7 which might cause or contribute to the disabling or recovering of plant system(s) during or following an accident.

The submittal does not clearly describe the types of human event considered for each post-initiator human event identified. For example, a human event identified may be the failure to feed and bleed, while the types of human errors considered may involve failure to follow procedures for feed and bleed, failure to open the correct valve (error of omission), or opening incorrect valve (error of commission).

Please list and describe in detail all the post-initiator human events considered in the IPE, including recovery actions and the 36 events in Table 3.3-7 and describe what types of human errors were considered for each human event.

20. Table 3.3-7 of the submittal briefly describes a total of 36 different post-accident operator and maintenance action events along with their associated HEPs, time windows (T_w) and PRA codes. Table 3.3-7 does not include $T_{1/2}$ and T_a for each event. The submittal is not clear on the

values of $T_{1/2}$ and T_a, how they were estimated and how they were used with the time-reliability correlation of EPRI NP-6937 for the various post-initiator human events.

Please identify and provide a description including HEP, T_w[1], $T_{1/2}$[2], T_a[3] and type (either mitigate consequences of, or recovery from, an accident) for all post-initiator human events considered in the IPE along with all the accident sequences they are contained in. Given the definitions for these times it is not clear how $T_{1/2}$ is used. Please provide an explanation of its use and its relationship with times T_w and T_{w-cr}. For the HEPs where expert judgment was used to develop the time-reliability correlation, please describe the process used to solicit and incorporate this judgment.

21. It is not clear from the submittal whether or not plant-specific performance shaping factors (PSFs) were used to modify human error probabilities (HEPs) and what the process was for reducing HEPs through the application of plant-specific PSFs. The plant-specific information could include the size of crew, availability of procedures, time available and time required, etc. The process could include examination of procedures, training, human engineering, staffing, communication, and administrative controls.

    Please a) provide a list of the types of plant-specific PSFs considered and their values, and b) discuss by way of examples how these PSFs were used to modify the HEPs. In addition, please provide a discussion of the process used to determine the appropriateness of applying PSFs to post-initiator human events.

22. Subsection 3.3.3.3, "Post-Accident Human Interactions," of the submittal has a segment entitled "Modeling Human Interaction Dependencies." The segment states that: "Each human action identified and evaluated in the system models, was reviewed in the content of the accident sequences to which it would contribute, to determine if there were actions appearing in those sequences which may not be totally independent.

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1 T_w: system time window - time between cue which initiates a given event and time when event must be complete for success

2 $T_{1/2}$: human response time - median time from cue to response by human

3 T_a: task implementation time - time it takes to complete event and therefore,

T_{w-cr}: time available for cognitive response (detection-diagnosis-decision making); $T_{w-cr} = T_w - T_a$.

It is not clear from the submittal if the identification of all the dependencies was addressed and how they were treated in the post-initiator HRA. The performance of the operator is both dependent on the accident under progression and past performance of the operator during the accident of concern. Improper treatment of these dependencies can result in the elimination of potentially dominant accident sequences, and therefore, the identification of significant events.

Please provide a list of all dependent events treated in the post-initiator HRA, showing the events that they were dependent upon.

INDIAN POINT 2 BACK-END QUESTIONS

23. Containment Configuration and Containment Phenomena -- Containment failure modes are discussed in Section 4.4.1 of the IPE submittal. Some containment phenomena (e.g., ex-vessel steam explosion, blowdown forces, and some hydrogen issues) are briefly discussed and dismissed because the NUREG-1150 Surry study shows them to have a small effect on containment integrity. Since the challenges (from loading conditions) to containment integrity depend on containment structural configurations, a more detailed examination and discussion is requested to identify plant-specific vulnerabilities.

    The containment configurations that are of interest to the above issues are the reactor cavity and the passage ways from the cavity to the upper containment. Detailed cavity design and the passage ways from the cavity to the upper containment are not discussed in the IPE submittal. The drawings provided in the submittal are small and crowded and hard to read, and no simplified drawings for containment features (e.g., cavity design, containment compartmentalization, etc.) are provided.

    Please supply legible, simplified drawings showing the IP2 cavity design and the passage ways from the cavity to the upper containment. Highlight the difference between IP2 and the reference plants used for comparison in your submittal, i.e., Surry and Zion. Explain why the reference plant results for ex-vessel steam explosion, high pressure melt ejection (HPME), and hydrogen combustion are applicable to IP2 considering the differences in geometry.

24. Power Recovery for Station Blackout (SBO) Sequences -- AC power recovery for SBO sequences after core damage is considered in the Level 2 analysis in the plant damage state (PDS) definition (Section 3.1.6.3.4). According to Figure 3.1-11, the probability of power recovery from offsite or the gas turbines subsequent to core damage but prior to vessel failure (about 1 to 2 hours) is 82.8%, the probability for power recovery from vessel failure to containment failure (about 12 hours) is 8.5%, and the probability of no recovery is 8.7%. No justification is provided in the submittal for the use of the above values. Please describe what operator actions are involved and how these values are obtained.

25. RCS Pressure -- One heading for the logic diagram used in the IPE submittal for PDS definition is the "RCS pressure at core damage and at vessel failure." This treatment of the RCS pressure seems to ignore RCS depressurization between these two times. Later discussion in CET development shows that hot leg failure is considered and will result in RCS depressurization. The use of this heading seems to cause some confusion and inconsistency in containment event tree (CET) quantification.

    In the CET quantification, induced primary system failure (Heading 1 of CET) is evaluated. However, according to the logic rules used for CET quantification, its effect on RCS pressure is not considered in the evaluation of early containment failure (Heading 3 of CET, CF-EARLY) using the Decomposition Event Tree (DET) (Page 4-182 for the logic rules for the DET in Figure 4.6-3 of the submittal). On the other hand, its effect on RCS pressure is considered in the evaluation of CHR and CS availability (HR-EARLY, Figure 4.6-4) and ex-vessel debris coolability (due to debris dispersing, EXVCOOL, Figure 4.6-5). Since induced RCS failure is assumed in the IPE to be about 72% and 3.4% for high-high and high RCS pressure, respectively, its effects are important. Please explain this apparent discrepancy in the CET quantification.

26. Hydrogen Combustion and CPI Recommendations -- The CPI recommendation for PWRs with a dry containment is the evaluation of containment and equipment vulnerabilities to localized hydrogen combustion and the need for improvements (including accident management procedures). This issue is not specifically addressed in the IPE submittal. Although effects of hydrogen combustion on containment integrity and equipment are discussed in Section 4.4.1 (on combustion processes) and 4.6.3 (on containment loading and failure of recirculation spray pumps and containment fans by environmental conditions), the discussions are brief. Detailed cavity design and the pathways between the cavity and the upper containment and their effects on combustion processes are not provided. (See Question 1)

    Since local hydrogen concentration depends on containment compartmentalization and the pathways between the cavity and the upper containment volume, please explain how the CPI issue was addressed.

27. Containment Isolation Failure -- Isolation failure is assigned a probability of 5E-4. This value is comparable to that used in the Surry NUREG-1150 study (2E-4), but much smaller than that used in Zion NUREG-1150 study (5E-3). The smaller value used in Surry is partly attributed to the subatmosphere containment condition. In this regard, IP2 is closer to Zion (both have large dry containment). It is stated in the backend of the IPE submittal (Section 4.4.1) that the value is based on Level 1 analysis. However, no discussion of containment

isolation is found in the Level 1 part of the IPE submittal, and containment isolation is not included in Section 3.2, System Analysis, of the IPE submittal. Please provide a discussion on containment isolation along the lines of Section 2.2.2.5 of NUREG-1335.

28. Harsh Environmental Conditions and Equipment Survivability -- The effects of harsh environmental conditions on the availability of containment sprays and fan coolers are discussed and quantified under CET headings 4 and 7 for early and late time frames (HR-EARLY and RS-LATE), respectively. The values used in the IPE depend heavily on engineering judgment. Potential severe accident conditions are discussed briefly. However, possible pressure and temperature time histories during severe accidents are not presented in the discussion. Since MAAP code calculations have been performed for the IP2, how were they factored into the quantification of these CET headings? If they were not used in their quantification, why not?

    The failure probability for recirculation pumps and fan coolers due to environmental conditions inside the containment was assigned a value of 0.1 (page 4-49). However, a value of 0.05 was used in later quantification (pages 4-54 through 4.57). Please explain the discrepancy and the basis for these values.

29. Early Containment Failure Grouping in the Sensitivity Study -- The early containment failure category used in the IPE for impact evaluation includes all containment failure modes that result in early releases, i.e., containment bypass, containment isolation failure, and early containment failure due to containment loading (e.g., DCH). Since early containment failure due to pressure load contributes only a small fraction (0.13% of total CDF) to this broader definition of early containment failure category (6.4% of total CDF, mostly from bypass failure), the evaluation of the impact as presented in the IPE submittal does not reveal the real impact on containment performance by uncertainties in RCS and containment phenomena (e.g., DCH, induced RCS depressurization, in-vessel debris cooling). In order to gain some insights on containment performance from uncertainties in RCS and containment phenomena, please provide the results of your sensitivity calculations for early containment failure exclusive of bypass and isolation failure.

30. Sensitivity to Debris Cooled Ex-Vessel and the Probability for Basemat Melt-Through Failure

    a) According to the IPE submittal, Late Failures (potential basemat melt-throughs) increased noticeably (from 1% to 13% of total CDF) if ex-vessel core debris was assumed not coolable. Another parameter that is important in the determination of long term containment failure (by basemat melt-through) is CF-LONG (containment failure long term given core debris not coolable ex-vessel) in the CET. The combination of these two parameters determines the probability of long term containment failure. CF-LONG was assumed in the IP2 IPE to have a value of 0.25 for a non-coolable debris pool. This was quoted in the submittal as based on the value used in NUREG/CR-4551. However, it is not specified in the submittal whether the value was based on the Surry (basaltic concrete) study or the Zion (limestone) study. From the value used in the submittal, it seems this was based on Surry data. In addition no discussion was provided for comparison of cavity configuration, concrete type, and basemat thickness. Please provide the basis for choosing the value of the parameter CF-LONG. (If Surry data is being used, then explain why it is applicable to IP2 which uses limestone.)

b) Many of the models used in the IPE Level 2 analysis are similar to those used in NUREG-1150; however, the ex-vessel debris coolability model used in the IP2 IPE is different from that used in NUREG-1150. The model used in the IPE considers in more detail, the debris dispersed out of the cavity, the depth of the debris pool, and the associated probabilities for debris cooled ex-vessel for the various conditions. However, the data used for the quantification of these parameters are primarily based on the engineering judgment of the analysts. It seems that the model used in the IPE would lead to a higher probability of debris cooling and, consequently, a lower probability of basemat melt-through failure. Please provide an estimate of the increase in the basemat melt-through failure mode for IP2 if one would use the NUREG-1150 modelling assumptions instead of those used in the IPE.

31. Clarification

a) In the sensitivity study of the recirculation spray failure probability, it is stated in the IPE submittal that this is investigated by the RS-Early DET (p4-94 of the submittal). However, there is no RS-Early DET in the DETs presented in the IPE submittal (Figures 4.6-1 through 4.6-9). Should it be the HR-Early DET? It is also noted that the heading addressing the initial availability of containment recirculation sprays is RECSPRAYS. Please clarify.

b) The probabilities for 'No Late RS Failure' and 'Yes - RS Failure Late' for Case C of page 4-76 are assigned values of 0.4 and 0.6, respectively. Should they be 0.883 and 0.113, respectively? If not, please explain this apparent discrepancy.

cc:

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