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NOV 06 1972

Docket No. 50-286

Consolidated Edison Company
of New York, Inc.
ATTN: Mr. William J. Cahill, Jr.
Vice President
4 Irving Place
New York, New York 10003

Gentlemen:

As a result of our continuing review of your application for an operating license for the Indian Point Nuclear Generating Unit No. 3, we find that we need additional information to complete our evaluation. The specific information required is listed in the enclosure.

In order to maintain our licensing review schedule we will need a completely adequate response by January 19, 1973. Please inform us within seven (7) days after receipt of this letter of your confirmation of the schedule or the date you will be able to meet. If you cannot meet our specified date or if your reply is not fully responsive to our requests, it is highly likely that the overall schedule for completing the licensing review for this project will have to be extended. Since reassignment of the staff's efforts will require completion of the new assignment prior to returning to this project, the extent of extension will most likely be greater than the extent of delay in your response.

Please contact us if you desire any discussion or clarification of the material requested.

Sincerely,

Original signed by
D. B. Vassallo

D. B. Vassallo, Chief
Pressurized Water Reactors
Branch No. 1
Directorate of Licensing

Enclosure:
Request for Additional Information

cc:
See page 2

OFFICE ▶			PWR-1	PWR-1	~~XXXXXXX~~	
SURNAME ▶			HSpecter:cls	DBVassallo		
DATE ▶			11/2/72	11/6/72		

cc w/encl:
LeBoeuf, Lamb, Leiby & MacRae
ATTN: Arvin E. Upton, Esq.
1821 Jefferson Street, N. W.
Washington, D. C. 20036

OFFICE ▶						
SURNAME ▶						
DATE ▶						

ADDITIONAL INFORMATION REQUIRED
INDIAN POINT NUCLEAR GENERATING
UNIT NO. 3
DOCKET NO. 50-286

1.0 General

1.4 Descibe the considerations for creation of secondary missiles on all buildings outside the containment following outside impact by the critical tornado missile. Evaluate only those building walls, roof slabs or domes which house seismic Category I equipment. Delineate the path of possible secondary missiles created from this event, list the seismic Category I systems which could be damaged or made inoperable, and the precautions taken, if required.

2.0 Site and Environment

2.6 Provide the following information on foundation conditions beneath Category I structures:

2.6.1 A plot plan showing the locations of all Unit 3 Category I structures, and the locations of borings, profiles, trenches and excavations.

2.6.2 Geologic profiles showing the relationship between Category I structures and the details of the foundation materials.

2.6.3 Identification and evaluation of deformational zones such as shears, joints, fractures, zones of structural weakness relative to Category I foundations.

2.6.4 Logs of borings not included in the PSAR or FSAR for all three reactor units, geologic logs of any trenches dug during the investigations, or logs of excavations for structures.

2.6.5 Any additonal evidence to support the conclusion that there are no cavities or cavernous conditions at the site.

2.6.6 Definition of site ground water conditions in light of any new information acquired since the PSAR review.

2.6.7 Geophysical data such as seismic refraction and uphole.

2.6.8 Any available site information to indicate the adequacy of design, such as foundation performance records since construction or settlement records.

2.6.9 A summary of the static and dynamic properties of foundation materials substantiated with representative laboratory test records.

2.6.10 A description of techniques and the adequacy of operations to improve foundation conditions, such as grouting, dental work, or rock bolting.

2.7 In maximizing hydrologic parameters for probable maximum flood (PMF) determinations, the assumption is generally made that an antecedent storm about half as severe as the PMF has occurred 3-6 days before the start of PMF precipitation. This assumption usually is sufficient to assume ground wetness, resulting in minimum losses and maximum rainfall excess. This was done satisfactorily in the FSAR. However, the antecedent storm is generally also sufficient to fill a substantial portion of the available flood control storage before substantial PMF runoff can occur. In this regard, justify the antecedent reservoir storage conditions assumed.

2.8 The verification of selected unit hydrographs presented in Section 2.5 of the FSAR is adequate. However, the routing coefficients should also be verified at selected locations by similar reconstitution methods where data are available.

2.9 Evaluate the coincidental wave action at the plant site using techniques presented in U.S. Army Coastal Engineering Center Technical Report No. 4, or a similar document. Also, for critical waterfront locations, determine the significant and maximum wave heights, and corresponding runup.

2.10 Since the occurrence of a PMF and a spring high tide may be postulated almost as readily as the three tide conditions presented for the Battery in Figure V-1 of the FSAR, provide the estimated PMF water level at the site concurrently with a spring high tide. In addition, describe what provisions have been made to account for the variable tidal flow between the Battery and the site. Also, further clarify the discharges used to compute the profiles shown in Figure V-1 of the FSAR.

2.11 The computations of surge attenuation effects are highly dependent on the selection of empirical coefficients. The number of historical surges in the Hudson, some of which are illustrated in Figure A-46 of the FSAR, would provide ideal data for coefficient verification. Accordingly, substantiate the surge attenuation coefficients by reconstituting at least one of the higher historical surges.

2.12 Provide a description of your current onsite meteorological program. Include parameters measured, types and characteristics of instruments, height of instruments, periods of data record and data recovery information.

2.13 Provide joint frequency distributions of wind direction and wind speed by atmospheric stability class using the ΔT method for a representative one (or preferably two) year period of record. Safety Guide 23 should be used for guidance on location of meteorological measurements and data stratification to be presented. Specify the percentage of data recovery for the period of record. Where periods of missing data are of days duration (as opposed to sporadic duration of a few hours) specify the periods of missing data. Present evidence as to the degree of representativeness of the period of record.

3.0 Reactor Design

3.14 Complete the response to our request No. 3.13 by verifying that the reactor internals for Indian Point Nuclear Generating Unit 3 will be subjected to a preoperational functional test program in accordance with the requirements of AEC Safety Guide 20, "Vibration Measurements on Reactor Internals," for similar plants.

3.15 With respect to Section 14.3.3 of the FSAR, "Primary System Pipe Rupture - Mechanical Integrity Analysis," provide your design basis for the combined seismic and LOCA effects on the fuel assembly or reference Westinghouse Topical Report WCAP-7950, "Fuel Assembly Safety Analysis for Combined Seismic and LOCA," dated July 1972, if applicable.

3.16 Westinghouse Topical Report WCAP-7332-L, "Indian Point No. 2, Reactor Internals Mechanical Analysis for Blowdown Excitation" is referenced in Section 14.3.3 of the FSAR. With respect to the no-loss-of-function deformation limits for the guide tubes adjacent to a ruptured outlet nozzle, this report apparently does not include the effect of axial loads. These loads could be significant due to a sudden decrease in pressure in the upper plenum following a rupture in the outlet nozzle. Provide a summary of the deformations and stresses in the guide tubes due to combined axial and lateral loads. Include a stability analysis of the guide tube, if appropriate. As an alternative, demonstrate that the axial loads are negligible.

3.17 The allowable stress criteria presented in Section 14.3.3 of the FSAR are applicable only to elastic analyses. Verify that elastic system dynamic analyses and elastic component analyses were used for the reactor internals under blowdown and seismic excitation.

4.0 Reactor Coolant System

4.28 Complete the response to our request No. 4.16 by revising Table 4.1-8 of the FSAR to reflect the number of design earthquake cycles. Describe the procedures which were used to account for the number of earthquake cycles during one seismic event, and specify the number of loading cycles for which Category I systems and components were designed.

4.29 Verify that the stress limits presented in Table A.1-2 of the FSAR were used only in conjunction with elastic system dynamic analyses and elastic component analyses.

4.30 Clarify the response to our request No. 4.18 by providing a summary of the dynamic analyses performed for Class I (seismic) piping and associated supports which determine the resulting loadings as a result of a postulated pipe break. Include the following:

a. The locations and number of design basis breaks on which the dynamic analyses are based.

b. The postulated rupture orientation, such as a circumferential and/or longitudinal break(s), for each postulated design basis break location.

c. A description of the forcing functions to be used for the pipe whip dynamic analyses. Include direction, rise time, magnitude, duration and initial conditions that adequately represent the jet stream dynamics and the system pressure differences.

d. Typical diagrams of the mathematical models used for dynamic analysis.

e. A summary of the analyses performed to demonstrate that unrestrained motion of ruptured lines will not sever adjacent impacted piping or pierce impacted areas of containment steel wall.

f. A description of the typical pipe whip restraints and a summary of number and location of all restraints used in each system.

4.31 The response to our request No. 4.17 in Amendment 21 is not satisfactory. An operational test program for all Class I (seismic) piping is required on all plants. Provide a description of the operational test program that will be used to verify that Class I (seismic) piping and piping restraints within the reactor coolant pressure boundary have been designed to withstand dynamic effects resulting from transient conditions. Identify the specific transients, pump trips, valve closures, etc. that will be performed during these tests. Include a description of the acceptance criteria (e.g., acceptable amplitude of vibration) that will be used during the test program to confirm the structural design of the Class I (seismic) piping and piping components.

4.32 The response to our request No. 5.20 indicates that floor response spectra were employed in the design of equipment and piping. Provide the details of the derivation of the floor response spectra.

5.0 Structures

5.34 With respect to the response to our request Nos. 5.17 and 5.20.2, provide justification for widening the peaks of the response spectra by only $\pm$ 8.5% of the period scale instead of the normally accepted widening of $\pm$ 10%.

5.35 Supplement the response to our request No. 5.19 by providing justification for the criteria used for the design of piping routed from one building to another to adequately account for the differential movement at support points. Include a description of the design employed to accommodate this differential movement.

5.36 The responses to our request Nos. 5.16, 5.21 and 5.25 state that static loads equivalent to the peak of the floor spectrum curves are used for the seismic design of non-rigid components and equipment. Justify the use of peak spectrum values by demonstrating that the contribution of all significant dynamic modes of response under seismic excitation have been included.

5.37 Clarify the response to our request No. 5.28 by providing justification for the criteria for combining modal responses (shears, stresses, moments, deflections and/or accelerations) where modal frequencies are closely spaced and a response spectrum modal analysis method is used.

5.38 Supplement the response to our request No. 5.13 by discussing the use of peak recorders to determine the response of selected Category I components and equipment under earthquake loading. Include a discussion of the plan that will be used to compare the measured component responses obtained from these recorders during an earthquake with the responses obtained from the seismic analysis of the components and equipment.

5.39 The answer to our request No. 5.1 and to the referenced request No. 5.18 does not cover the case of symmetrical buildings under seismic torsion. Indicate whether this effect was considered in the design of Class I (seismic) structures and, if omitted, justify the omission.

5.40 Indicate the extent to which AEC Safety Guide No. 18, "Structural Acceptance Test for Concrete Primary Reactor Containments," will be followed. If the testing program does not meet this guide, discuss the basis for concluding that your testing program is acceptable.

5.41 If the spent fuel racks are considered to be Class I (seismic), describe the design criteria and the analytical methods used in their design. State the design code that has been used and whether the maximum stresses are below the allowables for all loading conditions. State the criteria that have been used to establish the allowable strains. (See also question 9.14.9.)

6.0 Engineered Safety Features

6.16 Provide a P&I drawing of the post-accident containment venting system.

6.17 Safety Guide 7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," includes a statement that "...reactors should have the installed capability for a controlled purge of the containment atmosphere through appropriate fission product removal systems." Discuss how this will be accomplished in the Indian Point 3 plant, and provide the results of an analysis of the radiological consequences of purging the containment.

6.18 Discuss the bases, from a functional standpoint, for the selection of containment isolation valves including valve type, actuator, and closure time, and the required level of reliability.

6.19 Section 6.1.1 of the FSAR identifies the AEC General Design Criteria common to all engineered safety features and discusses how plant design satisfies each criterion. The discussion accompanying General Design Criterion 4 (see p. 6.1-7 of the FSAR) addresses the dual functions of plant systems and components, whereas General Design Criterion 4 is concerned with the sharing of systems or components between reactor facilities. Therefore, discuss the extent of sharing of Indian Point Unit 3 containment systems or components with the other reactor facilities at the Indian Point site.

6.20 Specify the ordinates for the NPSH curves in Figures 6.2-2, 6.2-3 and 6.2-4 of the FSAR.

6.21 Discuss in more detail the function of the timer associated with the containment spray system and state the delay time involved.

6.22 With respect to the carbon filter high temperature detection and dousing systems (containment air recirculation cooling and filtration system), provide the following information:

a. Specify the number of temperature switches provided in each carbon filter assembly;

b. Provide a drawing of a carbon filter assembly showing the distribution of the temperature switches and the arrangement of the dousing system nozzles; and

c. Discuss the provisions for filtering the water supplied to the dousing system from the containment spray system to prevent nozzle clogging.

6.23 Discuss the design provisions to assure that the air supply ductwork of the containment air recirculation cooling and filtration system remains intact following a postulated design basis loss-of-coolant accident.

6.24 Provide a P&I drawing of the post-accident containment atmosphere sampling system.

6.25 Based on the parameter values listed in Table 1 of Safety Guide 7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," provide an analysis of hydrogen production and accumulation in the containment following the design basis accident. Include in your analysis the effect of galvanized metal corrosion.

7.0 Instrumentation and Control

7.19 The response to our request No. 7.5.1 appears to be inconsistent with the cable separation criteria presented in FSAR Section 8.4 for cables installed in the electrical tunnels. Figure Q7.5.1 of the FSAR shows that 480 V power cables energized from buses 5A and 6A are installed in the lower tunnel with cables energized from buses 2A and 3A. Explain how this arrangement complies with your separation criteria.

7.20 Apparently there is a misunderstanding regarding the information requested in our request No. 7.11. Provide a description of the instrumentation available to the operator for monitoring conditions in the reactor, reactor coolant system, and containment. This description should address the number of instrument channels provided and the range, accuracy, and location of the indicators (such as meters and recorders). The analysis of design adequacy should address the margin between the ranges of indicators and recorders and expected variations of the monitored parameters in the event of an accident.

7.21 With reference to the response to our request No. 7.16:

a. Describe the extent to which the pressure interlock meets the requirements of IEEE Std 279-1971.

b. Clarify whether or not the pressure interlock is also used to close the valves to prevent overpressurization of the RHR system.

7.22 Expand the response to our request No. 8.8 to clarify whether or not all of the fuel storage tanks discussed are independent of the storage tanks discussed in the Unit 2 FSAR. Provide a diagram of the fuel oil storage and transfer system for the Unit 3 diesel generators and indicate any interconnections with other fuel oil systems.

7.23 Provide the design criteria for the instrumentation, control, and power systems associated with the auxiliary feedwater system.

7.24 It is noted that the normal power supply to instrument Bus 33 is disconnected as part of the load sequencing scheme. Identify, in more specific detail than that provided in Figure 8.2-5 of the FSAR, the function served by each individual load powered from Instrument Bus 33. Discuss how this arrangement complies with the requirements of Section 4.20 of IEEE-279 regarding anomalous indications and alarms. Provide an analysis to show that no single failure coincident with loss of power to Instrument Bus 33 will prevent any protective action.

7.25 Table 8.2-1 of the FSAR indicates that several motors will be loaded above their horsepower ratings. Describe the tests that have been performed to assure that these motors can withstand loading greater than their ratings.

7.26 It is noted that the containment cooling fan motors provided for Unit 2 have a rating of 350 hp. Discuss your reasons for concluding that 225 hp motors are adequate for Unit 3.

9.0 Auxiliary and Emergency Systems

9.23 Provide the basis for why the resin retention screens in the Chemical and Volume Control System (CVCS) mixed-bed and cation-bed demineralizers are not designed to withstand full system differential pressure for the fully clogged screen. Cite the reasons why they are only designed for a differential pressure of 25 psi, rather than full system pressure. Cite the safety considerations evaluated in the selection.

9.24 Your response to our request No. 9.14.1 in Supplement 8 states that the design tornado missiles will not penetrate the walls of the spent fuel pit. Also, the response states that should a missile hit the surface on the spent fuel pit water, by the time it reached the top of the fuel assemblies, its velocity would be reduced so that it would not damage the spent fuel.

Reevaluate the response, citing design tornado missiles striking the fuel storage building (refer to Appendix A, Section 3.2, regarding siding panels on the fuel storage building). Reevaluate those same missiles contacting fuel pit water surface, damaging spent fuel in the pit, and the plans to resolve the situation and control the release of radioactivity.

9.25 Your response to our request No. 9.14.2 in Supplement 8 gives a description of the spent fuel storage rack bracing to achieve seismic stability. Since the racks should be designed as seismic Category I equipment, classify the present design, and provide the basis regarding acceptablity of this design. Discuss whether the storage rack can withstand an uplift force and, if so, provide the magnitude of the force assuming it is applied to one fuel element location.

9.26 Your response to our request No. 9.14.3 in Supplement 8 notes that operation of the crane will be performed by qualified personnel. Discuss the plans, and program for assuring performance of qualified personnel. Compare your requirements to Chapter 2-3 "Operation - Overhead and Gantry Cranes, USAS B30.2.0-1967" as developed by the American National Safety Code for Cranes, Derricks, Hoists, Jacks and Slings.

9.27 Your response to our request No. 9.14.7 in Supplement 8 requires a more in-depth evaluation. The following additional information is required:

(a) For the missile shield structure above the reactor vessel, provide:

1. A sketch of the missile shield including its weight, dimensions, method of attachment to its handling fixture;

2. A sketch of the handling fixture and the relationship of its peroperational proof test load to its design and operating loads; and

3. The maximum drop height of the missile shield and a description of the means employed to limit it to the drop height noted above.

(b) For the reactor vessel head, provide:

1. A sketch of the reactor vessel head assembly showing its weight and method of attachment to its handling fixture:

2. A sketch of the reactor vessel closure head handling fixture showing the method of attachment to the overhead crane; and

3. The maximum drop height and a description of the means employed to limit the maximum drop height.

(c) For the upper core barrel assembly, provide:

1. A sketch of the upper core barrel showing its dimensions, weights, construction and method of attachment to the handling fixture and the overhead crane. Include a sketch of the handling fixture; and

2. The maximum drop height, and a description of the means employed to limit the maximum drop height.

(d) Using conservative conditions in each of the above cases, provide the results of an evaluation of the following:

1. The magnitude of impact load, its effect on the reactor vessel's points of support and the possible consequences of disrupting the flow of coolant to and from the reactor vessel and refueling canal and the consequences of such failure; and

2. The possibility for failure of the water seal between the reactor vessel and the refueling canal structure and the consequences of such a failure.

9.28 Your response to our request No. 9.15.2 in Supplement 8 notes that for the Service Water System (SWS) "the design is not capable of withstanding the additional simultaneous occurrence of a design basis earthquake, as these components are designated Class III seismic components." Please clarify whether this applies only to the back-up service water pumps located in Units 1 and 2 discharge canal. FSAR Figure 9.6-1 shows this back-up system as Class I (seismic). Verify that in Appendix A, Section 2, page A1-4 (a page revised in Supplement 7), which delineates Indian Point 3 service water pumps and piping as Class I (seismic). In addition explain the meaning of the wording "to the extent that water is always available to the service water pumps," following delineation of the intake structure as a Class I structure. Is the structure Class I?

9.29 FSAR Figure 9.6-1 shows two six inch flow control valves in the ten inch common header from the Diesel Generator Jacket Water and Lube Oil Coolers. Both flow control valves discharge to a ten inch line. In the case of failure of one of the two control valves, adequate flow does not appear assured through the remaining single valve. Evaluate and describe the selection of this design arrangement. State the factors considered and provide the allowable flow rates for the cooling system and the corrective action to be taken.

9.30 FSAR page 9.6.2-3 states that portions of the Fire Protection System (FPS) could be isolated to mitigate single failures in the system. Discuss the means by which the operator becomes aware of the existence of FPS pipe breaks or leaks. Discuss how their location is determined and how appropriate valving is accomplished. Since this system is an extension of the Unit 1 FPS, discuss the dependence of the Unit 3 operator on information displayed on the Unit 1 control panels.

9.31 Considering the gravity of the situation insofar as offsite doses, and difficulties that would be encountered in attaining and maintaining a safe shutdown if an undetected or uncontrolled fire should occur in the switchgear rooms, cable tray rooms, cable vaults and tunnels, emergency diesel generator rooms or charcoal filter beds; present a balanced discussion of the merits and considerations to providing redundant and Category I fire protection systems to cover events such as:

(a) Failure of a single fire detection device;

(b) Failure of pipes due to seismic events, explosions, pressure, missiles, etc.;

(c) Failure of valves to operate; and

(d) Inaccessibility of the area for manual fire control.

9.32 Your response to our request No. 9.17 in Supplement 8 should have addressed itself to all tanks containing gas under pressure. This should have been interpreted as all such tanks located outside the containment. Accordingly, describe, with reference to elevation and arrangement drawings, how adjacent Category I equipment, e.g., piping, components, power supply, and controls, will continue to be operable when adjacent gas pressure tanks fail. Describe the presence of protective barriers, walls, and design considerations taken to minimize the potential for missiles of this type. As a guide to minimum conditions refer to Occupational Safety and Health Administration OSHA 29 CFR 1910 Subpart H - Hazardous Material Sections 1910.101 Compressed Gas, 1910.103 Hydrogen and 1910.104 Oxygen, Subpart M - Compressed Gas Cylinders, 1910.167 Safety Relief Devices for Compressed Gas Cylinders, 1910.168 Safety Relief Devices for Cargo and Portable Tanks Storing Compressed Gases, 1910.169 Air Receivers.

9.33 Clarify your response to our request No. 9.18 in Supplement 8 as to whether the tanks mentioned include all such tanks containing bladders or unbonded liners outside the containment building. In addition, provide the results of an evaluation that includes the probability of pieces of the broken bladder or liner entering the tank discharge lines with potential for clogging a valve, strainer, and/or pump suction. State how this type of failure is detected and the impact this would have on operation and safety.

9.34 Your response to our request No. 9.22 in Supplement 8 presented a flow diagram, Figure Q9.22-1, of the Containment, Primary Auxiliary, and Fuel Storage Building Ventilation System. Provide an evaluation to justify the absence of a charcoal filter in the primary auxiliary building discharge duct. Provide an analysis based on the anticipated radioactive leakage from activities served by this system.

10.0 Steam and Power Conversion System

10.19 Your response to our request No. 10.4 in Supplement 7 notes that the plant is designed as practical as possible to avoid steam release to the atmosphere. Specifically, main steam is routed in all normal instances to the main condensers except under the three instances noted in your response to our request Nos. 10.8 and 10.9 in Supplement 8 when condenser isolation occurs; namely:

(a) The existence of low condenser vacuum beyond an allowable setpoint,

(b) The circulating water pump for the particular condenser section not running, and

(c) Loss of load interlock.

Provide the basis and the results of an evaluation of the low vacuum setpoint and its relationship to condenser shell and tube design. Discuss the selection of the prohibit setpoint on steam bypass valve operation and provide an evaluation of partial steam flows on decreased vacuum less than the setpoint.

Also, explain the reasoning for the requirement of the loss of load interlock prohibit on steam bypass valve discharge to the main condenser.

10.20 Your response to our request No. 10.14 provided in Supplement 8 refers to Section 4.7 of the Technical Specifications. In-plant testing of the steam generator stop valve is specified at refueling intervals with the plant at cold shutdown. Observations following this type of test would verify timely operation of the signal, actuator, and valve seating, as well as the instruments which display their motion, and measure their steam flow. Justify the absence of valve testing under operating pressure and temperature at nominal steam flow conditions.

10.21 Describe, keying your narrative to elevation and arrangement drawings provided in the FSAR and any additional elevation and arrangement drawings necessary to explain the point, how Category I systems, piping, components, power supply, and controls, in all buildings outside the containment, will continue to be operable should adjacent non-Category I, non-safety classified equipment fail in a catastrophic manner. Consider examples of these failures as circulating water system rupture, condensate system rupture, and gross overflow or rupture of storage tanks containing water, lube oil, or fuel oil.

10.22 Your response to our request No. 10.18 in Supplement 8 gives the most critical location of cracks in the last stage of low pressure turbine wheel. Explain the effect and impact of the critical crack orientation. Describe how ultrasonic techniques can locate such a crack. Describe the limitations of the present day ultrasonic techniques in their ability to detect laminar, radial, and multidirectional indications and any problem of accessibility for inspection. Based upon your response to the foregoing, provide your plans regarding inspection, including ultrasonic inspection, of the last stage turbine wheel and rotor during turbine overhaul periods.

11.0 Waste Disposal and Radiation Protection

11.13 Supplement 2, dated September 1972, to the Environmental Report indicates in Section 14.2, an increase in the steam generator blowdown rate from approximately 1 to 50 gpm. Discuss the basis for this change, the capacity of the Unit 1 treatment system to process this stream along with similar streams from Units 1 and 2, and the effects of flashing the steam generator blowdown to the Unit 1 instead of the Unit 3 turbine condenser. Provide the information used to calculate the liquid and gaseous releases from the secondary loop including secondary coolant composition, decontamination factors and iodine partition coefficients. This should include the process flowsheet with equipment capacities, and cycle times for loading and regenerating the demineralizers.

Identify all sources of blowdown steam at the Indian Point site. Evaluate the combined effect of these sources on the environment.

12.0 Conduct of Operations

12.14 Revise Fig. 12.1-1 of your FSAR, or designate in some other manner, the number of persons that are or will be assigned to the positions indicated in Fig. 12.1-1.

12.15 Designate, for other than the operating shift, those positions for which you will require the incumbent to hold an AEC Senior Operator License, and for which unit (Indian Point 1, 2, or 3) he will hold the license.

12.16 Designate the specific succession to responsibility for the overall operation of the facility and for Indian Point 3 in the event of the absence of the Manager, Nuclear Generation Department.

12.17 Provide the following information in regard to your Radiation Contingency Plan:

a. Describe the responsibilities and duties of State and local agencies that will be part of your offsite support, and the criteria for determining the need for participation of these agencies.

b. Your means for determining the magnitude of the release of radioactive materials in the event of a release of radioactive material that might require offsite action.

c. Your provisions for drills to test your means of communication with offsite agencies that might be called upon for support in the event of an emergency.

d. The agreements reached with Federal, State and local authorities.

e. Your arrangements for transportation of injured or contaminated individuals to treatment facilities outside the site boundary.

f. Your provisions for training groups other than the plant staff that might be called upon in the event of an emergency.

12.18 The response to our request No. 12.9 is inadequate in that it does not provide sufficient detail to evaluate your Industrial Security Plan for Indian Point 3, nor does it describe those changes that must be made to your current plan to extend it to cover Unit No. 3. Supplement your response to our request No. 12.9 by providing your detailed Industrial Security Plan for Unit No. 3 in its entirety or by reference to the specific parts of the Unit No. 2 <u>in camera</u> proceedings applicable to Unit No. 3 and then providing the additional provisions that will be needed to expand the plan to cover Unit No. 3.

13.0 Initial Tests and Operation

13.5 Your initial testing program described in Table 13.3-1 does not indicate that you intend to conduct the following tests described in the AEC Guide for the Planning of Initial Startup Programs, December 7, 1970; (a) Loss of Flow, (b) Turbine Trip, (c) Generator Trip, (d) Shutdown from outside control room, (e) Loss of offsite power and (f) psuedo rod ejection test. Justify your reasons for omitting these tests from your program or revise your program to include these tests, indicating the power level at which each test is to be conducted.

13.6 Describe for the WEDCO position of shift Startup Engineers, Chart II, Fig. 13.4-3 the following: Primary function, duties and responsibilities, and position requirements.

14.0 Safety Analysis

14.5 Since river water will be circulated through the containment fan coolers and since they will be used under both normal and accident conditions, discuss your plans for periodically verifying that the fan cooler heat removal capability does not degrade below that assumed in the containment integrity evaluation.

14.6 Provide the information listed below regarding the analysis of containment pressure transients. Explain all assumptions used in the analysis. Assumptions should be conservative with respect to the calculation of containment pressures.

a. For the spectrum of reactor coolant system pipe ruptures considered in the containment pressure transient analyses, specify the assumed locations of the postulated breaks. Include containment pressure transient analyses of various postulated loss-of-coolant accidents. A double-ended break of the largest reactor coolant outlet pipe and double-ended breaks of the reactor coolant pump suction and discharge pipe should be included. Smaller pipe breaks should also be analyzed and should be selected to be representative of the spectrum of break sizes for both inlet and outlet reactor coolant pipes. Assume only the minimum engineered safeguards are available to reduce the containment pressure; consider all delays in bringing the system into operation, e.g., the time to reach the containment pressure actuation signal, the delay time for equipment activation, and the time it takes the system to deliver rated flow to the containment. The analyses should be extended through the blowdown, reflood and post-reflood phases of the accidents.

b. Discuss in detail the calculational model that is used to describe the core reflood phase of a loss-of-coolant accident following initial blowdown. Include a discussion of the method used to calculate post-blowdown steam production and the steam venting rate to the containment, the assumed energy sources (such as core stored and decay energy, thick and thin metal stored energy, and steam generator stored energy) and the manner in which these energy sources are factored into the analysis.

14.7 Provide the following information with regard to the containment internal structures differential pressure analyses:

a. With respect to the analytical model used to predict the pressure buildup within a compartment, specify the time steps used in the analyses;

b. Discuss the results of the analyses performed for each compartment, including the maximum absolute and differential pressures attained, and the magnitude of the jet forces on the compartment walls; and

c. Discuss the structural design capability of each compartment to withstand the differential pressure and jet forces resulting from postulated loss-of-coolant accidents.

14.8 Listed below is the information that is needed to allow us to perform an independent assessment of the containment pressure transient analyses. Provide the following information:

a. The normal temperature of the water in the refueling water storage tank.

b. A curve of fan cooler performance showing energy removal rate as a function of containment atmosphere temperature.

c. The heat transfer area of a residual heat exchanger.

d. The average temperature of the primary coolant water under normal operating conditions.

e. A table of mass release to the containment (lb/sec) and enthalpy of the mass (Btu/lb) as functions of time throughout the blowdown and reflood phases of the postulated loss-of-coolant accidents resulting in the highest calculated containment pressures for both the hot leg and cold leg.

f. A table of mass release to the containment (lb/sec) and enthalpy of the mass (Btu/lb) as a function of time throughout the postulated steam line break accident.

g. Curves of the structural heat transfer coefficient as a function of time for the loss-of-coolant accidents identified in item (e) and the steam line break accident identified in item (f).

14.9 Supplement 7, page Q2.4-2 provides population data on state parks, military establishments and recreation areas. Provide similar data on schools, hospitals, and prisons within ten miles of the site. Indicate also variations in population on a seasonal basis and where appropriate variations in population distribution during the working day should be discussed, particularly where significant shifts in population or population distribution may occur within the low population zone (Figure 2.4 of the FSAR).

14.10 Provide a map showing all current industries within a five mile radius of the reactor site. Indicate the recreational use of the Hudson River in the vicinity of the Indian Point site. Provide a map(s) showing all current military bases, missile sites, munition storage areas, chemical plants and storage facilities, transportation routes (land and water), oil and gas pipelines, tank farms and military firing ranges.

14.11 Our evaluation of the consequences of the loss-of-coolant accident, based on Safety Guide No. 4 meteorological conditions (onsite data are still under staff review), indicates that the offsite doses are calculated in excess of the guideline values of 10 CFR Part 100. This was brought to your attention during our initial meeting pertaining to the operating license review for Unit No. 3. Describe what means will be taken to reduce these doses to acceptable levels.

14.12 In supplement 6, page 14.3-1-22 of the FSAR, it is indicated that F_q^T will be changed to 2.40 in the appropriate place in the technical specifications, in keeping with the assumptions used in the LOCA analysis. We presume the analysis of F_q^T to achieve this low value will be $F_z^N = 1.55$, $F_{xy} = 1.435$, $F_u^N = 1.05$ and $F_E = 1.03$. For operation with $F_q^T < 2.5$, we require frequent in-core surveillance of the reduced components of F_q^N. For the values listed, this would imply surveillance of F_z^N. Indicate how you intend to maintain $F_q^T \leq 2.4$ and your surveillance provisions.

14.13 With respect to the control room ventilation system, provide the following:

14.13.1 Describe the physical location of the fresh air inlets to the control room.

14.13.2 Describe the charcoal filter unit planned for use in the control room ventilation system.

14.13.3 Are all miscellaneous ducting and openings leading from and to the control room isolated during emergencies to eliminate possible inleakage, e.g., locker room exhaust duct?

14.13.4 The 35 cfm filtered air make-up assumed in the safety analysis does not agree with the design fresh air make-up of 1000 cfm reported in Section 9.0. The amount of air make-up should be based on maintaining at least a 1/8" Wg pressure within the control room. Using the 1000 cfm flow rate, our calculations indicate that the requirements of Criterion 19 are not met. Discuss the feasibility of accurate damper adjustment to achieve very low air-makeup flow rates. The tests proposed to verify that the control structure can be pressurized to 1/8" Wg at the flow rate claimed should be indicated.

14.13.5 *For two-inch charcoal bed filters a 90% iodine removal efficiency is normally applied for elemental iodine and a 70% removal efficiency for methyl iodine. How does this affect the thyroid dose presented in the FSAR?

14.13.6 *Present the beta dose for the Design Basis Loss of Coolant Accident computed for control room operators. Discuss any departures from Safety Guide 4 calculational methods. Discuss any actions proposed to reduce beta exposure of personnel.

14.13.7 How will the ventilation system be operated (using the three modes discussed in Section 9.9.3) during the first few hours after the Design Basis Adcident as compared to the first few days?

14.13.8 Describe the initial verification test program and periodic surveillance tests necessary to assure system availability and proper operation.

14.13.9 Present a calculation of the doses received by control room operator from a main steam line break accident.

*In the case of the LOCA, allowances may be made for control room occupancy. Occupancy factors of 1.0 for 0 - 24 hours, 0.6 for 1 - 4 days, and 0.4 for 4 - 30 days are acceptable.

APPENDIX B QUALITY ASSURANCE

B2.0 The description of the QA Program (alternatively known as the QA Program Plan) in the FSAR describes the QA Program for the design and construction phase, but provides insufficient information on how the Consolidated Edison Company (CE) will implement each of the 18 QA criteria of Appendix B to 10 CFR 50 during the post-construction phases and over the life of the plant. Accordingly, a description of the QA Program for Operations should be provided. Each element within each criterion of Appendix B to 10 CFR 50 should be addressed and discussed relative to all post-construction activities for this project, including preoperational testing, initial fuel loading, initial plant operation, and full term operational efforts.

B2.1 Provide organization charts and describe the organizational arrangement denoting the role of CE's QA and QC personnel during all post-construction phase efforts and over the life of the plant. Include a description of how and when the QA/QC personnel will interface with other CE home office and site organizations. Describe whether there is an interface role between CE's QA/QC organization and the various committees utilized for facility operation. Organizational charts should clearly denote lines and areas of communication, responsibility, and authority.

B2.2 Describe for CE's QA Program for Operations those parts of ANSI N45.2 and draft ANS 3.2 that CE intends to follow toward fulfillment of the Regulatory requirements provided in Appendix B to 10 CFR 50. Cross reference may be made between Appendix B to 10 CFR 50 and the provisions within ANSI N45.2 and draft ANS 3.2.

B2.3 Provide a list of the titles of the QA/QC documents contained in CE's QA Manual for Operations. Briefly summarize the purpose and content of each of these documents. Where CE reports exist to provide such response these may be referenced in the description of CE's QA Manual for Operations.

B2.4 Describe CE's system for the preparation, review, approval, revision, distribution, and control of the QA Manual for Operations. Indicate how it is assured that the appropriate departments and organizations will properly implement these documents.

B2.5 Describe the responsibility of CE's QA/QC personnel with respect to the review, approval, and implementation of changes to QA/QC documents, test procedures, procurement documents and plant operating and maintenance procedures. Also describe their role with respect to the following activities: plant testing, routine surveillance, repair, replacement, calibration, training, inservice inspection programs, and independent audit.

B2.6 Indicate the projected number and location of QA/QC personnel required and assigned by CE during all post-construction phases and during the life of the plant.

B2.7 Describe the qualifications and training requirements for personnel in CE's QA/QC organization.

B3. Describe CE's System for communicating information concerning abnormal experiences at other facilities, including AEC's Reactor Operations Experience Reports and Reactor Construction Reports to the appropriate design, construction, and operating organizations, and for assuring that the experiences embodied in these reports are considered in the program efforts.

B.4 Describe CE's policy and system for implementing AEC's Codes and Standards Rule and Deficiencies Reporting Rule.

B5. Describe training programs for QA/QC personnel. Describe whether there are any orientation and training programs to familiarize other CE headquarters and site personnel with the requirements of Appendix B to 10 CFR 50.

B6. With regard to Criterion XI of Appendix B to 10 CFR 50, "Test Control," what involvement do CE's QA/QC personnel have with the planning, procedures, and implementation of plant tests?

B7. Describe the calibration policy, schedule, and system planned for plant operations to meet the requirements of Criterion XII of Appendix B to 10 CFR 50 "Control of Measuring and Test Equipment." What is the role of the QA/QC personnel relative to the calibration program? Include both portable and installed instrumentation and equipment utilized in calibration, as well as a discussion of plant equipment subject to periodic calibration. Include a list of references of titles to industry or government calibration standards and/or specifications that will be invoked as part of CE's calibration program.

B8. With regard to "Inspection, Testing and Operating Status":

a. Describe CE's tagging and other measures to be invoked for meeting Criterion XIV of Appendix B to 10 CFR 50. Describe the responsibilities of CE's QA/QC staff and other CE personnel with respect to this activity.

b. Describe the policy and system for logging and tagging the status of inoperative and malfunctioning components in such a manner that their status cannot be overlooked in operating the plant.

B9 Describe the role of CE's QA/QC staff during the operating phase with respect to Criterion XV of Appendix B to 10 CFR 50 "Nonconforming Material, Parts, and Components." What mechanism exists to assure timely notification of all affected parties for those cases where repair, rework, and/or reduction of requirements are anticipated? Describe the policies and steps established to assure that appropriate organizations evaluate discrepant and unacceptable materials or components and decide proper disposition. Describe the organizational arrangements for evaluation, the membership and duties of review boards (if there are to be such), and the level of management which is to be made cognizant of the actions taken in this area.

B10. With regard to Criterion XVIII of Appendix B to 10 CFR 50 "Audits", describe the nature and extent of the audit program planned for plant testing, fuel loading, initial startup, operations, maintenance, refueling, future purchase of material and services, and inservice inspection efforts. Describe the estimated frequency of audits over various home office and plant activities, and describe those audits which are to be performed by Committees versus those to be performed by CE's QA/QC personnel. Describe the audit reporting policy, follow-up responsibility, and provisions for review of audit reports by top management of CE.

B11. With regard to the deficiencies noted in the letter of September 25, 1972, from J. P. O'Reilly of AEC Regulatory Operations to W. W. Lapsley of Consolidated Edison Company, provide a discussion of the documented procedures and implementing activities, including management controls now in effect that correct and preclude repetition of these deficiencies. Although these deficiencies are directly applicable to the Quality Assurance Program for operations for Indian Point Units 1 and 2, these findings appear to also be applicable to the description of the Quality Assurance Program for operations for Indian Point Nuclear Generating Unit No. 3.