4

JUL 24 1970

Docket No. 50-247

Distribution:
Docket File ✓
AEC PDR
DR Reading
DRL Reading
PWR #1 Reading
C.K. Beck, DR
M.M. Mann, DR
F. Schroeder, DRL
R. S. Boyd, DRL
R. C. DeYoung, DRL
D. Skovholt, DRL
T.R. Wilson, DRL
E.G. Case, DRS
R. R. Maccary, DRS
R. Klecker, DRL
B/C's, DRL & DRS
N. Blunt, DRL
T. Engelhardt, OGC
ACRS (18)
W. Nyer (2)
N.M. Newmark

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

Gentlemen:

At a subcommittee meeting of the Advisory Committee on Reactor Safeguards held on May 28, 1970 to discuss your application for an operating license for the Indian Point Nuclear Generating Unit No. 2 the ability to accurately predict the response of a reactor core upon initiation of emergency core cooling was discussed. This area of concern is of sufficient importance to warrant documentation of the status of the information available at this time for the Indian Point facility. Accordingly, please reply to the requests for additional information listed in Attachment 1 to this letter.

In addition, in the course of our review we have noted a number of other areas in which the information presented in the Final Facility Description and Safety Analysis report is either incomplete or is not consistent with our present understanding of your intent. We have listed the areas in which clarification is requested in Attachment 2 to this letter. The areas have been numbered sequentially as a continuation of our first and second requests for additional information issued on August 4, 1969, and November 13, 1969, respectively.

Please call if you desire clarification of any of these questions.

Sincerely,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosures:
1. Attachment 1
2. Attachment 2

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

OFFICE ▶	DRL:PWR #1	DRL:PWR #1	DRL:AD/PWR's	~~DRL:D DIR~~	DRL:DIR
SURNAME ▶	Kniel/sp	Muller	DeYoung	~~Schroeder~~	Morris
DATE ▶	7/24/70	7/24/70	7/24/70	~~7/ /70~~	7/24/70

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

July 24, 1970

Docket No. 50-247

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

Gentlemen:

At a subcommittee meeting of the Advisory Committee on Reactor Safeguards held on May 28, 1970 to discuss your application for an operating license for the Indian Point Nuclear Generating Unit No. 2 the ability to accurately predict the response of a reactor core upon initiation of emergency core cooling was discussed. This area of concern is of sufficient importance to warrant documentation of the status of the information available at this time for the Indian Point facility. Accordingly, please reply to the requests for additional information listed in Attachment 1 to this letter.

In addition, in the course of our review we have noted a number of other areas in which the information presented in the Final Facility Description and Safety Analysis report is either incomplete or is not consistent with our present understanding of your intent. We have listed the areas in which clarification is requested in Attachment 2 to this letter. The areas have been numbered sequentially as a continuation of our first and second requests for additional information issued on August 4, 1969, and November 13, 1969, respectively.

Please call if you desire clarification of any of these questions.

Sincerely,

Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosures:
1. Attachment 1
2. Attachment 2

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

ATTACHMENT 1

14.0 Accident Analysis

Your analysis of the thermal response of the core following a loss-of-coolant accident, as documented in your Final Facility Description and Safety Analysis Report, is based upon an analysis using the FLASH computer code to predict core parameters. We understand that a considerable amount of new information on the core response has been developed with the use of multinode analytical techniques, including the SATAN computer code. Please submit a comprehensive status report of the applicable information obtained using these new analytical techniques. This report should include information in response to the following specific requests:

14.12 Provide the results of your evaluation of the LOCA, using a multinode analytical model (such as the SATAN code) for a 27 1/2 inch ID, double-ended, cold-leg pipe rupture. In addition to providing information on clad temperatures and system pressures, also provide the core and hot channel flow rates in sufficient detail to fully characterize the thermal and hydraulic performance during blowdown. These details should include:

(a) core pressure drop, quality, mass velocity;
(b) hot channel pressure drop, quality, mass velocity;
(c) heat flux distribution in hot channel;
(d) flow rates in upper and lower plenums;
(e) flow rate in broken and intact cold-leg and hot-leg piping; and
(f) flow rate out the break.

Identify the heat transfer correlations used for the various phases of the blowdown and refill period and relate these correlations to the most recent experimental data available.

14.13 In the same manner, provide the results of your evaluation of a 29 inch ID, double-ended, hot-leg pipe rupture.

14.14 Provide evaluations, using your latest analytical techniques, to determine:

(a) the limiting loss-of-coolant break size for the rupture of (i) a cold-leg pipe, and (ii) a hot-leg pipe, for which assured core cooling is predicted;

(b) the limiting reduced power level for which assured core cooling is predicted, for the double-ended rupture of (i) a cold-leg pipe, and (ii) a hot-leg pipe, and

(c) estimates of the volumes of core associated with local areas of potential flow instability.

14.15 Provide a summary discussion regarding your acceptance criteria for the ECCS functional performance. Your discussions should include:

(a) identification of any supporting information which has become available as a result of the Commission-sponsored emergency core cooling test programs, and

(b) an assessment of the adequacy of your analytical techniques, including the SATAN code, to accurately predict core behavior during the loss of coolant accident. A discussion should be presented on each area of uncertainty, with an estimate of the probability for more or less adverse consequences than those predicted.

ATTACHMENT 2

1.0 GENERAL

1.13 With reference to the seismic design of the piping supports:

(a) State the magnitude and effects of out-of-phase seismic loads, and

(b) Verify that the piping systems have been analyzed based on "as built" support locations.

4.0 REACTOR COOLANT SYSTEM

4.12 Document the design and operating requirements for the equipment that will be used to detect failed fuel during plant operation.

7.0 INSTRUMENTATION AND CONTROL

7.20 For some relatively small breaks in the primary system the only signal available to initiate scram would be the low pressurizer pressure signal. As we have discussed with you previously, it would be desirable to add a diverse scram signal which would assure timely reactor scram in the event that the pressurizer signal fails. Please state your intent in this regard and provide a supporting analysis to demonstrate that the core will be protected over the range of break sizes which would require this scram signal to shutdown the reactor.

7.21 Provide a description of the instrumentation that will be available to the plant operator and the procedures he will use to assess the course of each of the postulated accident conditions.

7.22 Discuss how both the control and power circuits for the boron injection tank valves meet the IEEE 279 criteria for both opening and closing actions.

7.23 Submit the procedures for the testing of the initiating and control instrumentation for Engineered Safety Features.

7.24 Document that scram breaker "position" lights will be added in the control room to alert the operator as to the position of the scram breakers.

7.25 The response to question 7.1 indicates that the reactor trip on turbine trip, and the turbine runback circuits meet IEEE 279. It was subsequently determined that these circuits do not meet IEEE 279 and that they need not meet it since they are anticipatory signals and are not required for reactor safety. Change the response to question 7.1, accordingly.

7.26 Correct Page 8.2-14 of the FSAR to delete the mention of automatic switching of the bus tie breakers between the vital buses.

7.27 Complete your documentation on the seismic testing of protection system equipment. WCAP 7397-L, "Seismic Testing of Electrical and Control Equipment" does not include all electrical equipment necessary to the operation of the protection system.

7.28 We understand that you intend to make the manual actuation of the containment spray independent of the automatic portion of the circuit. Please describe your intent in this regard.

8.0 ELECTRICAL SYSTEMS

8.5 Discuss the analysis performed to determine that additional restraints are required for the instrument air line which passes near the 480 volt essential switchgear. Further, describe the barrier which will be installed to shield the switchgear and cables from potential missiles that could originate from the air compressor.

8.6 Describe the additional work being performed in the electrical penetration area to provide added assurance of cable protection.

8.7 Describe the concrete wall which will be installed to shield the diesel generator control panel from potential missiles that could originate from the diesel generators.

8.8 Describe the equipment which would sense undervoltage of the essential buses, and signal that the diesel generators be started.

12.0 CONDUCT OF OPERATIONS

12.8 Provide personnel resumés for the Superintendent Performance, Supervisor Engineering (Health Physics), Assistant Superintendent (Maintenance), Assistant Supervisor Engineering (Nuclear Plant Instrumentation, Health Physics and Conventional Plant Instrumentation) and the remaining General Watch Foremen.

12.9 Indicate, relative to Figure 1, Section 12, Supplement No. 2, the anticipated number of individuals under the following job titles; Maintenance Mechanics, Technical Assistants (Chemist), Senior Production Technicians (Shift Chemist), Production Technicians (Chemist), Senior Production Technicians, Production Technicians (Performance), Technicians (Nuclear Plant Instruments), Technicians (Shift Health Physics) and Technicians (Conventional Plant Instruments).

12.10 Indicate on Figures 1 & 2, Section 12, Supplement No. 2, all positions for which you intend to license personnel on Unit No. 2; whether the licenses are Senior Operator Licenses or Operator Licenses; and whether these persons will be "cold" or "hot" licensed.

12.11 Has the staff of the Superintendent Performance and/or the staff of the Supervisor Engineering (Health Physics) been expanded for Unit No. 2 operation and if so, describe the specific training received by the new personnel, including course content and number of hours. Describe the training to be received by the Superintendent Performance, Assistant Superintendent (Maintenance) and the Supervisor Engineering (Health Physics), including course content and the number of hours.

14.0 ACCIDENT ANALYSIS

14.6 Based on calculations supplied in response to Question 14.2 (Supplement No. 8) the radiation doses that would be received by personnel in the plant control room following a design basis loss-of-coolant accident do not meet the criterion currently required for approval of construction permit applications. This criterion requires that exposures be limited to 5 Rem whole body, or its equivalent, to any part of the body for the duration of the accident. Although we may not require absolute conformance to the dose criterion that we now apply to construction permit reviews, some modification of the design of the control room would be desirable so as to increase the assurance that the health and safety of the operating staff, and thus their efficiency and effectiveness in the event of an accident, would be protected in an acceptable manner. Accordingly, summarize those design modifications that could be made to reduce the radiation doses to approach those specified above using a spray inorganic removal coefficient of 4.5 hr^{-1} and a charcoal organic removal efficiency that is justified by present experimental data.

14.17 Document the type, manufacturer, and the flow characteristics of the chemical additive spray system nozzles used for post LOCA iodine removal.

14.18 Provide the design details of the charcoal adsorber system with respect to the type, weight, and distribution of charcoal in the filter units, and the arrangement of the filter units in each plenum.

14.19 With reference to the equipment to be installed in the containment vessel to handle post-accident radiolytic hydrogen, please provide the following information:

(a) Discuss more fully the suction arrangement of the air supply blower for the recombiner units and the connection with the ring distribution header from the recirculation cooling and filtration system. **Provide a sketch of the arrangement, which includes indications of the local air circulation patterns to the recombiner suction point and the specific locations** of the **sampling lines.**

(b) Discuss the location selected for introducing oxygen makeup into the containment and the means whereby mixing of the oxygen is assured.

(c) Higher than ambient hydrogen concentrations (arising from unmixed gases evolved within the containment or from leakage of recombiner fuel lines) might exist near a recombiner unit prior to its startup. It is not clear from our review of the information submitted to-date that such local conditions could be detected prior to a recombiner startup or is precluded by the unit design. Our concern relates to potential flame propagation upon recombiner startup and to local and unit damage that might result from such propagation. It appears that the ability to sample locally, prior to recombiner startup and perhaps periodically during its operation, could be of advantage to the safe operation of the recombiner unit. Provide a discussion of this matter and state the design provisions, such as local air circulation rates, and sampling capability that currently exist in your design to preclude or to detect higher than ambient hydrogen concentrations in the regions near the recombiner units.

(d) We understand that in contrast to your response to Question 6.8b(1) regarding the design of the post-accident containment sampling system, your plan is to employ a vacuum pump arrangement to provide sufficient flow in the sampling lines. Describe the sampling system design that will be installed. Support your description with suitable drawings.

(e) Your response to Question 6.8b(4) and (5) is not clear with respect to the post-installation testing and test frequency, and with respect to the processing setpoints you intend to establish for the recombiner units. Provide more specific information on the combustor and the diluent air flow settings and on the oxygen depletion range within which you would expect to operate.

(f) We note that in order to permit "throttle-back" operation of the recombiner unit, bypasses will need to be used or certain adjustments in the protective devices for the flame failure system may need to be performed at the external control station. Describe the actions that will be required to permit "throttle-back" operation including the types of bypass or adjustment actions that will need to be taken. State whether these operations will be simulated in the periodic testing programs.

14.20 Provide a discussion of the potential for, and consequences of, missiles generated by failure of any of the main turbine-generator units planned to be operated at the site.