William J. Cahill, Jr.
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June 12, 1980

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

Director of Nuclear Reactor Regulation
ATTN: Mr. Darrell G. Eisenhut, Director
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Eisenhut:

Attachment A to this letter contains our response to your May 7, 1980 letter regarding five additional TMI-2 related requirements. Your May 7, 1980 letter also indicated that guidance regarding the implementation of Item 1, Shift Manning, would be forwarded under separate correspondence. We have not as yet received that guidance. We will respond to Item 1 as expeditously as possible after receipt of your guidance.

Very truly yours,

William J. Cahill, Jr.

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Vice President

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ATTACHMENT A

Response to

"FIVE ADDITIONAL TMI-2 RELATED
REQUIREMENTS TO OPERATING PLANTS"
May 7, 1980

Consolidated Edison Company of New York, Inc.

Indian Point Unit No. 2

Docket No. 50-247

June, 1980

REVISED SCOPE AND CRITERIA FOR LICENSING EXAMINATIONS (I.A.3.1)

POSITION

Licensee's training program for reactor operators is to be upgraded to meet the new criteria as specified in the Commission's letter of March 28, 1980.

RESPONSE

We will comply.

PROCEDURES FOR FEEDBACK OF OPERATING EXPERIENCE TO PLANT STAFF (I.C.5)

POSITION

In accordance with Task Action Plan I.C.5, Procedures for Feedback of Operating Experience to Plant Staff, each licensee shall review its procedures and revise them as necessary to assure that operating information pertinent to plant safety originating both within and outside the utility organization is continually supplied to operators and other personnel and is incorporated into training and retraining programs. These procedures shall:

(1) Clearly identify organizational responsibilities for review of operating experience, the feedback of pertinent information to operators and other personnel and the incorporation of such information into training and retraining programs;

(2) Identify the administrative and technical review steps necessary in translating recommendations by the operating experience assessment group into plant actions (e.g., changes to procedures; operating orders);

(3) Identify the recipients of various categories of information from operating experience (e.g., Supervisory personnel, STA's, operators, maintenance personnel, H. P. technicians) or otherwise provide means through which such information can be readily related to the job functions of the recipients.

(4) Provide means to assure that affected personnel become aware of and understand information of sufficient importance that should not wait for emphasis through routine training and retraining programs;

(5) Assure that plant personnel do not routinely receive extraneous and unimportant information on operating experience in such volume that it would obscure priority information or otherwise detract from overall job performance and proficiency;

(6) Provide suitable checks to assure that conflicting or contradictory information is not conveyed to operators and other personnel until resolution is reached; and,

(7) Provide periodic internal audit to assure that the feedback program functions effectively at all levels.

RESPONSE

We will comply. Appropriate changes to the Indian Point No. 2 procedures will be completed by January 1, 1981.

INSTALLATION AND TESTING OF AUTOMATIC PORV ISOLATION SYSTEM (II.K.3.1)

POSITION

(a) All PWR licensees should provide a system which uses the PORV block valve to protect against a small break LOCA. This system will automatically cause the block valve to close when the reactor coolant system pressure decays after the PORV has opened, to relieve excess pressure. An override feature should be incorporated. Justification should be provided to assure that failure of this system would not decrease overall safety by intensifying plant transients and accidents.

(b) Each licensee should perform a confirmatory test of the automatic block valve closure system installed in response to (a) above.

RESPONSE

We do not believe an automatic PORV isolation system should be required. This is based on Westinghouse Owners Group analyses of the ultimate heat sink function, and the decreased intensity of a number of plant transients, given the PORVs operation. Failure of the proposed automatic PORV isolation system could impair this function. In addition, the plant modifications, procedure changes, and operator training (e.g., NUREG-0578 requirements) provide assurance that the function of the automatic isolation system will be provided by operator action. In addition, failure to isolate stuck open PORVs has been analyzed and results in no core uncovery.

PWR VENDOR REPORT ON PORV FAILURE REDUCTION (II.K.3.2)

POSITION

(a) Each PWR vendor should submit a report for staff review documenting the various actions which have been taken to decrease the probability of a small break LOCA caused by a stuck-open PORV and show how they constitute sufficient improvements in reactor safety. This report should be submitted for staff review.

(b) Safety valve failure rate based on past history of the vendor designed operating plants should be included in the report submitted in response to (a) above.

RESPONSE

A report on PORV failure reduction will be submitted to the NRC by January 1, 1981. It is currently anticipated that this report will be in the form of a generic Westinghouse Owners Group submittal.

REPORTING SAFETY AND RELIEF VALVE FAILURES AND CHALLENGES (II.K.3.3)

POSITION

(a) Future failures of a relief valve to close should be reported promptly to the NRC.

(b) Future challenges to the relief valves should be documented in the annual report.

(c) Future failures of a safety valve to close should be reported promptly to the NRC.

(d) Future challenges to the safety valves should be documented in the annual report.

RESPONSE

Future failures of pressurizer relief or safety valves to close will be reported promptly to the NRC in accordance with existing Technical Specification requirements.

The present Technical Specifications for Indian Point No. 2 requires monthly operating reports in lieu of an annual report. Therefore, challenges of pressurizer relief or safety valves will be reported to the NRC in the Monthly Operating Report.

AUTOMATIC TRIP OF REACTOR COOLANT PUMPS DURING LOCA (II.K.3.5)

POSITION

Tripping of the reactor coolant pumps in case of a LOCA is not an ideal solution. The licensees should consider other solutions to the small break LOCA problem (for example, an increase in safety injection flow rate). In the meantime, until a better solution is found, the reactor coolant pumps should be tripped automatically in case of a small break LOCA. The signals designated to initiate the pump trip should be carefully selected in order to differentiate between a small break LOCA and other events which do not require reactor coolant pump trip as discussed in NUREG-0623.

RESPONSE

In response to your IE Bulletin 79-06C (August 30, 1979 letter from W.J. Cahill, Jr. to Boyce H. Grier) we indicated the Westinghouse Owners Group expected to submit a report on reactor coolant pump trip to the NRC by August 31, 1979. This report (WCAP-9584) was submitted to the NRC on August 31, 1979 (letter from Cordell Reed, Chairman Westinghouse Owners Group, to John Stolz). This WCAP is the basis for the Westinghouse and Owners Group position on RCP trip (i.e., automatic RCP trip is not necessary for a Westinghouse PWR since sufficient time is available for manual tripping of the RCPs). This philosophy has been incorporated in the Westinghouse Emergency Operating Instructions which were reviewed and approved by the NRC Bulletins and Orders Task Force and subsequently incorporated in the Indian Point Unit No. 2

emergency operating procedures. The Westinghouse criteria (basically a RCS pressure below the shutoff head of SI pumps) provides for continued RCP operation and therefore forced circulation and decreased reliance on operator action for non-LOCA events. We have reviewed this report in conjunction with the Indian Point Unit No. 2 procedures. Based on this review it was determined no further revision of the operating procedures was required. As requested by the NRC in a letter dated April 15, 1980 and as discussed with the NRC during the May 22, 1980 meeting on this subject, we anticipate that the Westinghouse Owners Group will provide predictions of the LOFT test L3-6. The NRC has indicated that small break tests at the Semiscale and LOFT facilities as well as Owners Group test predictions will aid in NRC resolution of this issue. Therefore, we believe that it is not appropriate to take any additional actions on this issue until the results of the NRC sponsored testing (in particular L3-5 and L3-6) and Owners Group predictions are completed and the results evaluated.

PROPORTIONAL INTEGRAL DERIVATIVE (PID) CONTROLLER MODIFICATION (II.K.3.9)

POSITION

The Westinghouse-recommended modification to the Proportional Integral Derivative (PID) controller should be implemented by affected licensees.

RESPONSE

The initiation circuitry for the Pressurizer PORV high pressure trip does not include a Proportional Integral Derivative (PID) controller. Therefore, this position is not applicable to Indian Point No. 2.

PROPOSED ANTICIPATORY TRIP MODIFICATION (II.K.3.10)

POSITION

The anticipatory trip modification proposed by some licensees to confine the range of use to high power levels should not be made until it has been shown on a plant-by-plant basis that the small break LOCA probability resulting from a stuck-open power-operated relief valve (PORV) is little affected by the modification.

RESPONSE

Currently we are not proposing a modification of the anticipatory trip (i.e., turbine trip above 10% power results in a reactor trip). However, should such a modification be proposed in the future, it will be documented and an implementation schedule provided for NRC approval.

CONFIRM EXISTENCE OF ANTICIPATORY TRIP UPON TURBINE TRIP (II.K.3.12)

POSITION

Licensees with W-designed operating plants should confirm that their plants have an anticipatory reactor trip on turbine trip. The licensee of any plant where this trip is not present should provide a conceptual design and evaluation for the installation of this trip.

RESPONSE

Indian Point No. 2 has an anticipatory reactor trip on turbine trip.

REPORT ON OUTAGE OF ECC SYSTEMS - LICENSEE REPORT AND PROPOSED TECHNICAL SPECIFICATION CHANGES (II.K.3.17)

POSITION

Several components of the ECC systems are permitted by Technical Specifications to have substantial outage time (e.g., 72 hours for one diesel-generator; 14 days for the HPCI system). In addition, there are no cumulative outage time limitations for ECC systems. Licensees should submit a report detailing outage dates and lengths of outages for all ECC systems for the last five years of operation. The report should also include the causes of the outages (e.g., controller failure, spurious isolation).

RESPONSE

We will comply. A detailed report will be submitted by January 1, 1981.

REVISED SMALL-BREAK LOCA METHODS TO SHOW COMPLIANCE WITH 10 CFR 50, APPENDIX K (II.K.3.30)

POSITION

The analysis methods used by NSSS vendors and/or fuel suppliers for small break LOCA analysis for compliance with Appendix K to 10 CFR Part 50 should be revised, documented, and submitted for NRC approval. The revisions should account for comparisons with experimental data, including data from the LOFT and Semiscale facilities.

RESPONSE

The present Westinghouse small break evaluation model used to analyze Indian Point No. 2 is in conformance with 10CFR Part 50, Appendix K. However, Westinghouse has indicated that they will, nevertheless, address the specific NRC items contained in NUREG-0611 in a model change scheduled for completion by July 1, 1983.

PLANT SPECIFIC CALCULATIONS TO SHOW COMPLIANCE WITH 10 CFR 50.46 (II.K.3.31)

POSITION

Plant-specific calculations using NRC-approved models for small break LOCAs as described in II.K.3.30 above, to show compliance with 10 CFR 50.46 should be submitted for NRC approval by all licensees.

RESPONSE

The present Westinghouse small break evaluation model and small break LOCA analyses for Indian Point No. 2 are in conformance with 10CFR Part 50, Appendix K and 10CFR part 50.46. As noted in the response to item II.K.3.30, Westinghouse plans to submit a new Small Break Evaluation Model to the NRC for review by July 1, 1983. If the results of this new Westinghouse model (and subsequent NRC review and approval) indicate that the present small break LOCA analyses for Indian Point No. 2 are not in conformance with 10CFR 50.46, a new analysis utilizing the new and approved Westinghouse model will be submitted to the NRC in accordance with the NRC schedule.

CONTROL ROOM HABITABILITY REQUIREMENTS (III.D.3.4)

POSITION

In accordance with action item III.D.3.4, Control Room Habitability, licensees shall assure that control room operators will be adequately protected against the effects of accidental release of toxic and radioactive gases and that the nuclear power plant can be safety operated or shut down under design basis accident conditions (Criterion 19, "Control Room," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50).

RESPONSE

We will comply. A detailed report on Control Room habitability will be provided by January 1, 1981.