POWER AUTHORITY OF THE STATE OF NEW YORK

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June 16, 1980
IPN-80-58

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

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Five Additional TMI-2 Related Requirements

Dear Sir:

Attachment 1 to this letter provides the Authority's response to your May 7, 1980 letter regarding the subject requirements.

Please note that Items II.K.3.25, II.K.3.29 and II.K.3.44 have not been addressed in Attachment 1 based on notification from the NRC (Mr. L. Olshan) that these items are applicable to BWRs only. In addition, Mr. Olshan has informed the Authority that the correct implementation dates for Items II.K.3.30 and II.K.3.31 are those found in Enclosure 1 of the May 7, letter.

Very truly yours,

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Senior Vice President
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ATTACHMENT 1

RESPONSES TO

FIVE ADDITIONAL TMI-2 RELATED REQUIREMENTS

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
JUNE 16, 1980

I.A.3.1 Revised Scope and Criteria for Licensing Examinations

RESPONSE:

The Authority is presently implementing the requirements of the NRC March 28, 1980 letter in accordance with the schedule provided therein.

I.C.5 Procedures for Feedback of Operating Experience to Plant Staff

NRC 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:

The Authority will modify its procedures in accordance with the NRC I.C.5 position and will submit documentation of the method for NRC staff review by January 1, 1981.

II.K.3.1 Installation and Testing of Automatic PORV Isolation System

NRC 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 PORV(s) 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 PORV(s) has been analyzed and results in no core uncovery. Therefore, the Authority will take no further action at this time.

II.K.3.2 PWR Vendor Report on PORV Failure Reduction

NRC 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 a 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.

II.K.3.3 Reporting Safety and Relief Valve Failures and Challenges

NRC 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:

The Authority will promptly report failures of pressurizer safety and relief valves to the NRC in accordance with the IP3 Technical Specifications.

In addition, the Authority will provide a report of pressurizer safety and relief valve failures and challenges, since April 1, 1980, by January 1, 1981, and thereafter it will be incorporated into the annual report.

II.K.3.5 Automatic Trip of Reactor Coolant Pumps During LOCA

NRC 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 our response to IE Bulletin 79-06C dated September 10, 1979 (IP-WDH-5578), we referenced the Westinghouse Owners Group analysis of delayed RCP trip during small break LOCAs documented in WCAP-9584. 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 3 Emergency Operating Procedures. In addition 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. As requested by the NRC in a letter dated April 15, 1980 and as discussed with the NRC during the May 22, 1980 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.

II.K.3.9 Proportional Integral Derivative (PID) Controller Modification

NRC POSITION:

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

RESPONSE:

The Authority will implement the Westinghouse recommended modification to the PID controller by July 1, 1980. The derivative action has effectively been removed from the controller. In addition, the Authority will submit documentation of the method of removal to NRC by July 1, 1980.

II.K.3.10 Proposed Anticipatory Trip Modification

NRC 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:

To date, the Authority has not proposed 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 submitted to NRC for approval.

II.K.3.12 Confirm Existence of Anticipating Trip Upon Turbine Trip

NRC 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 Unit 3 has an anticipatory trip upon turbine trip. Therefore, no further action is required.

II.K.3.17 Report on Outage of ECC Systems - Licensee Report and Proposed Technical Specification Changes

NRC POSITION:

Several components of the ECC systems are permitted by Technical Specifications to have substantial outage times (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:

The Authority will submit a report in accordance with the NRC II.K.3.17 position by January 1, 1981. However, the report will only extend back to April 1976 which is the initial criticality date for Indian Point Unit 3.

II.K.3.30 Revised Small Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K

NRC POSITION:

The analysis methods used by NSSS vendors and/or fuel suppliers for small break LOCA analysis for complaince with Appendix K to 10 CFR 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 Unit 3 is in comformance 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.

II.K.3.31 Plant Specific Calculations to Show Compliance with 10 CFR 50.46

NRC POSITION:

Plant-specific calculations using NRC-approved models for small break LOCAs as described in II.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 Unit 3 are in conformance with 10CFR Part 50, Appendix K and 10 CFR Part 50.46. As stated 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 analysis for Indian Point Unit 3 are not in conformance with 10CFR Part 50.46, a new analysis utilizing the new and approved Westinghouse model will be submitted to the NRC in accordance with the NRC schedule.

III.D.3.4 Control Room Habitability Requirements

NRC 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 safely 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:

The Authority will provide a report in accordance with the NRC II.D.3.4 position and clarification by January 1, 1981. Required modifications, if any, will subsequently be scheduled for completion by January 1, 1983.