January 21, 1980

Docket No. 50-289

Mr. R. C. Arnold
Senior Vice President
Metropolitan Edison Company
100 Interpace Parkway
Parsippany, New Jersey 07054

Dear Mr. Arnold:

During the preparation of our safety evaluation covering the TMI-1 Restart Report, which was issued on January 11, 1980, certain positions and requests for additional informati n were identified by the staff. In the interest of expediting your respo es, these were conveyed informally to you in mid-December, 1979. Enclosure 1 documents these requests. Enclosure 2 identifies a clarification of a requirement in the safety evaluation.

Many of the items in Enclosures 1 and 2 reflect open issues in our safety evaluation. We understand that your staff has been preparing responses to many of these items. Within seven days of receipt of this letter, please inform us of your schedule for responding to the items in Enclosures 1 and 2 and to the open items in the safety evaluation.

Sincerely,

Original signed by
Richard H. Vollmer

Richard H. Vollmer, Director
Three Mile Island Support

Enclosures:
1. Request for Additional Information - TMI-1
2. Automatic Operation of PORV Block Valve

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Enclosure 1

REQUEST FOR ADDITIONAL INFORMATION

THREE MILE ISLAND, UNIT 1

1. Your response to questions 36 and 37 does not provide the staff with sufficient information to make an evaluation of the high pressure injection (HPI) design and associated flow rates. We require that you provide the following information:

   a. Table of expected HPI flow (1 and 2 HPI pumps) in each of the four legs versus RCS pressure (2500 to atmospheric) considering the new cavitating venturi installation. Provide your analytical/empirical basis for these flow rates. What reduction in flow rate was caused by the inclusion of these flow-limiting devices? Compare these flow rates to the HPI flow rates assumed in the B&W LOCA analysis, "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plants," May 7, 1979, Volume 1, Figure 6.2.59.

   b. A complete test description for confirmation of adequate flow splits and flows including:

      (1) description of temporary flow indications and where they will be installed. (Address why using installed instrumentation is not adequate.)

      (2) basis for 550 gpm "upper limit" acceptance criteria.

      (3) range of pressures over which data will be taken.

      (4) range of installed flow instrumentation.

      (5) acceptance criteria for flow rates at higher pressures.

   We require that this test confirm that the TMI-1 HPI design provides adequate flow as assumed in Figure 6.2.59 of the B&W analysis (above). Provide your commitment to conduct a test and submit the test procedure which will accomplish this purpose.

2. Your response to question 36a. does not provide sufficient analytical justification for adequacy of the 64/36 flow split for an HPI line break or your statement that RCS pressure will not expend significant time above 1500 psig for a spectrum of HPI line breaks. Provide such analyses or confirm that a 70/30 flow split would be achieved and that the existing LOCA analyses are appropriate for a spectrum of HPI line breaks (between the RCS and the check valve nearest the RCS).

3. Your response to question 36c. does not provide the staff with sufficient information to make an evaluation of the cavitating venturi design. Provide justification in the form of test data, calculations, etc., that the cavitating venturis can be relied upon to perform their function for an HPI line break (limit flow at break such that sufficient HPI reaches core). It is our position that the brief test description does not adequately cover the conditions which would result from an HPI line break. Also, provide detailed drawings, data, and specifications for the cavitating venturis.

4. Item 2.1.7.a of the Lessons Learned requirements states, in part, the following:

   The automatic initiation signals and circuits (of the Emergency Feedwater System) shall be designed so that a single failure will not result in the loss of system function.

   Further review of your proposed design for EFW system has brought into question the capability of the EFW flow control valves to meet the single failure criterion in the automatic mode. Our concern is based upon the non-single-failure-proof ICS as the sole source of automatic control signals to the two EFW flow control valves. (No credit can be taken for the manual control stations in your analysis.)

   Provide a detailed discussion of this aspect of your design that is responsive to the above concern. If conformance to the above requirement cannot be demonstrated, your response should also include the following: (1) a commitment to upgrade the design to meet this requirement on an expedited basis, (2) a proposed schedule for completion, and (3) a conceptual design of the proposed modification.

5. Lessons Learned, Item 2.1.8(a), Improved Post-Accident Sampling Capability

You stated in your TMI-1 Restart Report (Section 2.1.2.4, Amendment No. 4) that the design and operational review and conceptual design will be forwarded to the NRC for review by January 1, 1980.

You also stated that you will utilize reactor coolant system letdown monitors to meet this requirement and that these monitors are capable of remaining on scale with 10% failed fuel. For on-line monitoring, we will require the capability of post-accident sampling from a zone of the reactor coolant which is representative of in-core conditions. Since the letdown stream can be isolated in the event of an accident, we do not consider a sampling point in the letdown stream to be representative of in-core conditions. Further, we will require the range of on-line instrumentation to be capable of measuring coolant activity up to and including a release to the coolant of 100% of the core inventory of noble gases, 50% of the core inventory of halogens, and 1% of all other nuclides mixed in the reactor coolant. Therefore, the monitors you proposed do not meet our requirements.

6. Item 2.1.8(b), Increased Range of Radiation Monitors

Provide the following additional information:

4.1 For noble gas effluents

4.1.1 System/Method description including:

4.1.1.1 Instrumentation to be used including energy dependence, and calibration frequency and technique.

4.1.1.2 Monitoring/sampling locations, including methods to assure representative measurements and background radiation correction.

4.1.1.3 A description of method to be employed to facilitate access to radiation readings.

4.1.2 Procedures for conducting all aspects of the measurement/analysis including:

4.1.2.1 Procedures for minimizing occupational exposures.

4.1.2.2 Calculational methods for converting instrument readings to release rates based on exhaust air flow and taking into consideration radionuclide spectrum distribution as function of time after shutdown.

4.2 For radioiodine and particulates effluents

Procedures for conducting all aspects of the measurement analysis including:

4.2.1 Minimizing occupational exposure

4.2.2 Calculational methods for determining release rates

4.2.3 Procedures for dissemination of information

4.2.4 Calibration frequency and technique

4.3 A design review and installation schedule for TMI-1 reactor containment building radiation monitors.

7. Item 2.1.8(c), Improved In-Plant Iodine Monitoring Instrument

Provide the following additional information

5.1 Equipment type and model

5.2 Monitor range, readout modes, and calibration method

5.3 Associated training and procedures for accurately determining the airborne iodine concentration.

<u>SOLID RADWASTE SYSTEM (Section 7.3.1.3, Amendment No. 8)</u>

8. You state that the TMI-1 solid waste will be stored with EPICOR-II wastes until a permanent waste storage building is available. Justify the EPICOR-II waste staging area has enough capacity to accommodate the TMI-1 solid waste and provide in detail the description and availability of a permanent waste storage building you stated.

9. Provide the process control programs for temporary mobile solidification systems and permanent solidification system.

10. Provide the TMI-1 permanent solidification system capacity.

11. The long-term requirement of IE Bulletin 79-05C requires the B&W licensees to submit a design which will assure automatic tripping of the operating reactor coolant pumps (RCPs) under all circumstances in which this action may be needed. It has been shown through analysis that this trip is needed for a certain spectrum of small break LOCAs. To assure that the operating RCPs are tripped during LOCA conditions yet minimize the possibility of tripping the RCPs during non-LOCA transients (such as severe overcooling events), other B&W licensees have proposed a conceptual safety-grade trip circuit which would utilize coincident input signals of low pressure ESFAS actuation combined with low RCP current/power.

The use of these coincident signals was discussed in detail during a meeting held in Bethesda, Maryland on November 8, 1979, between the staff, the B&W Owners' Group and B&W. The staff expressed two concerns with the proposed design. First, while the staff agrees that a decrease in RCP current/power will occur if voiding takes place in the reactor coolant system, the relationship of RCP current/power versus void fraction has neither been fully documented by the licensees nor demonstrated through full scale testing. Secondly, absent the above information, the staff is not convinced that the proposed setpoint of 20% decrease in RCP current is the proper setpoint for that portion of the coincidence logic.

Although these matters must be resolved prior to approval of the final design, we believe that the proposed design contains sufficient flexibility and can be expected to meet its intended function. Therefore, B&W licensees should proceed with equipment procurement and development of a final design based on the proposed concept.

Prior to final design acceptability, the following conditions must be satisfied:

a. Characteristic curves for RCP current/power versus void fraction must be fully demonstrated and documented based upon existing test data and supplemented as necessary with confirmatory data obtained from future tests such as LOFT, full scale testing, etc.;

b. Justification for the RCP current/power setpoint must be shown; and,

c. Satisfactory responses to the following must be received.

In addition, a proposed schedule for installing the automatic RCP trip circuitry shall also be provided. Your installation schedule should show that this modification will be completed prior to reactor restart.

We have reviewed several preliminary design descriptions of the automatic trip circuitry for the subject motors which have been provided by most B&W plant licensees. Since it has been concluded that this action is required to perform a safety function, the added circuitry should conform to specified safety criteria.

When submitting the proposed final design for the automatic trip circuitry, provide responses to the following questions:

1. For your proposed design, state the degree of conformance with the applicable acceptance criteria listed in Column 7.2 of Table 7-1 of the Standard Review Plan. Also, provide justification for any non-conformance.

2. Provide a discussion of the following:

   a. conformance of the design with the design requirements of Section 4 of IEEE Std. 279-1971; and,

   b. conformance of the design with the principal design criteria of Section 4 of IEEE Std. 308-1974.

3. Provide a detailed description of any changes to and/or interfaces with the existing protection systems. Include diagrams (block, location, functional, and/or elementary wiring), as necessary, to clearly depict the changes and/or interfaces. In addition, provide an analysis which demonstrates that these changes and/or interfaces will not degrade the existing protection systems.

4. Identify equipment which is identical to equipment utilized in existing safety-grade systems. For the equipment which is not identical, briefly describe the differences.

5. In general, the equipment shall be seismically and environmentally qualified. Therefore, provide the following descriptive information for the qualification test programs:

   a. equipment design specification requirements;

   b. test plan;

   c. test setup;

   d. test procedures; and,

   e. acceptability goals and requirements.

When submitting your final design for approval, if all of the above information is not available, provide a schedule for its submittal.

6. Provide the criteria for the overall trip circuitry installation testing which will demonstrate that this circuitry has been installed properly. When submitting your final design for approval, if this information is not available, provide a schedule for its submittal.

12. By letters dated August 31, 1979, each B&W operating plant licensee indicated a general endorsement of B&W's generic report BAW-1564, "Integrated Control System Reliability Analysis."

Our joint review of this report with Oak Ridge National Laboratory has progressed sufficiently to assure ourselves that the recommendations that the report offers with regard to potential areas of improvement in ICS reliability are reasonable. Therefore, we request that you address these recommendations and discuss your followup action plans in this matter. Responses to the following items must be provided.

As part of the continuing review of this report, additional areas may be highlighted as requiring improvement. In that event, we will provide additional requests in these specific areas as necessary.

For the following recommendations from report BAW-1564,

- Describe how you plan to implement the recommendation; and
- The schedule for the implementation; or
- Basis for not implementing the recommendation at your plant.

a. ICS-Related

1. NNI/ICS power supply reliability

   NOTE: This area is of particular importance because it is listed as the source of most of the ICS input failures. The staff is not only concerned with the "reliability" of the power supplies, but also the design philosophy of the power supply implementation. It appears that power supply failures can lead to multiple problems such as the Rancho Seco event of March, 1978.

2. Reliability of input signal from the NI/RPS system to the ICS - specifically, the RC flow signal.

3. ICS/BOP system tuning, particularly feedwater condensate systems and the ICS controls.

   NOTE: Although this concern is related to tuning, it appears that more basic design and/or operational problems in the feedwater (and related) system may exist.

   Therefore, include a discussion of the following items:

   (a) Any particular operational (startup, etc.) problems experienced at your plant with respect to the ICS. Reference to previously submitted information is acceptable.

(b) Bases for operator intervention in place of automatic ICS action (including start-up, power operation and shutdown activities).

(c) Procedures used by the operator to perform the operation described in 1c(2) above.

(d) Additional training provided to the operator.

b. Balance of Plant

1. Main feedwater pump turbine drive minimum speed control - to prevent loss of main feedwater or indication of main feedwater.

2. A means to prevent or mitigate the consequences of a stuck-open main feedwater startup valve.

3. A means to prevent or mitigate the consequences of a stuck-open turbine bypass valve.

13. On page 6 of the Commission Order of August 9, 1979, Item 4 states:

> "The licensee shall demonstrate that decontamination and/or restoration operations at TMI-2 will not affect safe operations at TMI."

In addition to information already provided, you should address this requirement with respect to systems and operations other than those directly connected with waste management and effluent monitoring.

14. The licensee should address the corrective actions taken in response to the recommendations of the February, 1979, NUS Corp. audit of the TMI Radiation Protection program.

Enclosure 2

ATOMATIC OPE[ATION OF PORV BLOCK VALVE

1. In our safety evaluation, we noted in Order .tem 1, sub-item 1d, that we will require "that the PORV block volve be automatically closed on low pressure."

To clarify this requirement, a control system should be designed and installed to provide interaction between the PORV and block valve to prevent a small break LOCA in the event of a failure of the PORV to close. One such design would cause the block valve to close after the PORV opens on high pressure and subsequently the reactor coolant system pressure decays below the PORV reset pressure. This system would be provided with an override so that pressure relief could be accommodated at lower pressures, if required. In addition, the licensee should evaluate the overall effect of this control system on plant transients and accidents.

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Enclosure 2

ATOMATIC OPERATION OF PORV BLOCK VALVE

1. In our safety evaluation, we noted in Order Item 1, sub-item 1d, that we will require "that the PORV block volve be automatically closed on low pressure."

To clarify this requirement, a control system should be designed and installed to provide interaction between the PORV and block valve to prevent a small break LOCA in the event of a failure of the PORV to close. One such design would cause the block valve to close after the PORV opens on high pressure and subsequently the reactor coolant system pressure decays below the PORV reset pressure. This system would be provided with an override so that pressure relief could be accommodated at lower pressures, if required. In addition, the licensee should evaluate the overall effect of this control system on plant transients and accidents.

R. C. Arnold January 21, 1980

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