BOSTON EDISON COMPANY
GENERAL OFFICES 800 BOYLSTON STREET
BOSTON, MASSACHUSETTS 02199

G. CARL ANDOGNINI
SUPERINTENDENT
NUCLEAR OPERATIONS DEPARTMENT

November 21, 1979

BECo. Ltr. #79-241

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

License No. DPR-35
Docket No. 50-293

Response to Discussion of Lessons Learned
Short Term Requirements

References:
A. Followup to Reviews Regarding the Three Mile Island Unit 2 Accident, dated October 19, 1979

B. Discussion of Lessons Learned Short Term Requirements, dated October 30, 1979.

Dear Sir:

You have requested that all Operating Nuclear Power Plants provide a detailed description and justification for each of our responses to NUREG-0578 positions, submitted in Reference A, which are not in complete agreement with your Staff's requirements as clarified in Reference B.

Within the limited response time allowed by Reference B, we have performed a preliminary review of your requirement clarifications and have provided responses accordingly (2.1.8a, 2.1.8.b.3, 2.1.8.c, 2.1.9 New 1 and 2). Should continuing detailed review reveal further disagreements, we shall notify you as they arise and provide the requisite descriptions and justifications at that time.

Additionally, Boston Edison was contacted by Messrs. J. Hannon and D. Verrelli of your staff on November 14, 1979. They informed us that the NRC has completed their review of our technical positions taken in Reference A and that our positions had been accepted with five (5) exceptions. The exceptions were discussed and agreements reached on needed additional responses. Therefore, each of these identified positions, with its agreed response, is included in this letter.

We trust this letter is responsive to your requirements; however, should you desire additional information or clarification please feel free to contact us.

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Very truly yours,

G. Carl Andognini

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IMPROVED POST-ACCIDENT SAMPLING CAPABILITY (2.1.8.a)

POSITION

A design and operational review of the reactor coolant and containment atmosphere sampling systems shall be performed to determine the capability of personnel to promptly obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18 3/4 Rems to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safety obtain the samples, additional design features or shielding should be provided to meet the criteria.

A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify (less than 2 hours) certain radioisotopes that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperatures), and non-volatile isotopes (which indicate fuel melting). The initital reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (Regulatory Guide 1.3 or 1.4 source term). Both analyses shall be capable of being completed promptly; i.e., the boron sample analysis within an hour and the chloride sample analysis within a shift.

RESPONSE

A design review of the existing reactor coolant and containment atmosphere sampling stations, the spectrum analysis facilities, and the wet chemistry analysis facilities indicate that samples could not be collected or analyzed under the postulated accident conditions and stay within the defined personnel exposure criteria.

Conceptual design modifications are being developed by the BWR Owners' Group and the resultant proposals will be adapted for implementation by Boston Edison at Pilgrim Station. Justifications will be provided if this cannot be achieved on schedule.

1396 260

IMPROVED IN-PLANT IODINE INSTRUMENTATION UNDER ACCIDENT CONDITIONS (2.1.8.c)

POSITION

Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

RESPONSE

We take exception to the first recommendation, i.e., a portable cart with attached single channel analyzer. The detector shielding required to perform sample analysis in the presence of dose rates to be encountered under accident conditions would make movement of the cart extremely cumbersome and slow. This would cause unnecessary increased exposure to the person(s) collecting the sample. Also, due to the high, fluctuating detector background, the accuracy of the sample results could not be relied upon. In lieu of the above recommendation, we propose to use portable vacuum pump with a particulate filter and charcoal or silver zeolite cartridge. The samples would then be removed to a low background area for iodine analysis (thus, satisfying the 1-1-81 recommendation).

We also take exception to one part of the second recommendation. As stated above, all air samples for iodine would be removed to a low background, low contamination area for initial analysis, not <u>further</u> analysis. We feel the accuracy of the results and the reduced personnel exposure more than compensate for the time required to obtain the results.

We feel full implementation of improved in-plant iodine instrumentation under accident condition including associated training and procedures can be implemented before commencement of the next operating cycle.

PERFORMANCE TESTING FOR BWR AND PWR RELIEF AND SAFETY VALVES (2.1.2)

POSITION

Pressurized Water Reactor and Boiling Water Reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents.

RESPONSE

Boston Edison Company will adopt the GE Owners' Group position on safety and relief valve testing.

1396 262

DIRECT INDICATION OF POWER-OPERATED RELIEF VALVE AND SAFETY VALVE POSITION FOR PWRs AND BWRs (2.1.3.a)

POSITION

Reactor System relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve position detection device or a reliable indication of flow in the discharge pipe.

RESPONSE

BECo will adopt the GE Owners' Group position on posit:on indication for unpiped safety relief valves.

1396 263

CONTAINMENT ISOLATION (2.1.4)

POSITION

1. All containment isolation system designs shall comply with the recommendations of SRP 6.2.4, i.e., that there be diversity in the parameters sensed for the initiation of containment isolation.

2. All plants shall give careful reconsideration to the definition of essential and non-essential systems, shall identify each system determined to be essential, shall identify each system determined to be non-essential, shall describe the basis for selection of each essential system, shall modify their containment isolation designs accordingly, and shall report the results of the re-evaluation to NRC.

3. All non-essential systems shall be automatically isolated by the containment isolation signal.

4. The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action.

RESPONSE

Valve logic modifications will be completed before the beginning of our next operating cycle or justification will be provided if this cannot be achieved on this schedule.

1396 264

DEDICATED H_2 CONTROL PENETRATIONS (2.1.5.a)

POSITION

Plants using external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere should provide containment isolation systems for external recombiner or purge systems that are dedicated to that service only, that the redundancy and single failure requirements of General Design Criterion 54 and 56 of Appendix A to 10 CFR 50, and that are sized to satisfy the flow requirements of the recombiner or purge system.

RESPONSE

Dedicated penetrations for external recombiners will be installed before the beginning of our next operating cycle or justification will be provided if this cannot be achieved on this schedule.

1396 265

INCREASED RANGE OF RADIATION MONITORS (2.1.8.b.3)

POSITION

The requirements associated with this recommendation should be considered as advanced implementation of certain requirements to be included in a revision to Regulatory Guide 1.97, "Instrumentation to Follow the Course of an Accident", which has already been initiated, and in other Regulatory Guides, which will be promulgated in the near-term.

In-containment radiation level monitors with a maximum range of 10^8 rad/hr shall be installed. A minimum of two such monitors that are physically separated shall be provided. Monitors shall be designed and qualified to function in an accident environment.

RESPONSE

We concur with the position with one exception. The design will not comply in full with the requirements of IEEE-338 paragraph 6.3.4 (Standard Criteria for Periodic Testing of Nuclear Power Generating Station Safety Systems). BECo is investigating alternate methods to satisfy staff requirements. We shall provide NRC with design descriptions and justifications as they become available.

1396 266

CONTAINMENT PRESSURE INDICATION 2.1.9 NEW 1

POSITION

A continuous indication of containment pressure should be provided in the control room. Measurement and indication capability shall include three times the design pressure of the containment for concrete, four times the design pressure for steel, and minus five psig for all containments.

RESPONSE

Proposed Regulatory Guide 1.97, Rev. 02, (Instrumentation for Light-Water Cooled Nuclear Power Plant to Assess Plant Conditions During and Following an Accident), invokes IEEE-338, 1977 and IEEE-279, 1971. These standards are too limiting and can only be applied selectively to PNPS 1 in general, and specifically in relation to the additional containment level indication.

The system as will be designed will not meet the requirement for response time testing contained in paragraph 6.3.4 of IEEE-338.

The system will not meet the single failure criteria while in the test mode. Since this is a system strictly designed for display of information, and since the duration of testing of each loop will be short, this exception of single failure criteria is acceptable.

BECo is investigating alternate methods to satisfy staff requirements. We shall provide NRC with design descriptions and justifications as they become available.

1396 267

CONTAINMENT WATER LEVEL INDICATION 2.1.9 New 2

POSITION

A continuous indication of containment water level shall be provided in the control room for all plants. A narrow range instrument shall be provided for PWRs and cover the range from the bottom to the top of the containment sump. A wide range instrument shall also be provided for PWRs and shall cover the range from the bottom of the containment to the elevation equivalent to a 600,000 gallon capacity. For BWRs, a wide range instrument shall be provided and cover the range from the bottom to 5 feet above the normal water level of the suppression pool.

RESPONSE

Proposed Regulatory Guide 1.97, Rev. 02 (Instrumentation for Light-Water Cooled Nuclear Power Plant to Assess Plant Conditions During and Following an Accident), invokes IEEE-338, 1977 and IEEE-279, 1971. These standards are too limiting and can only be applied selectively to PNPS 1 in general, and specifically in relation to the additional containment level indication.

The system as will be designed will not meet the requirement for response time testing contained in paragraph 6.3.4 of IEEE-338.

The system will not meet the single failure criteria while in the test mode. Since this is a system strictly designed for display of information, and since the duration of testing of each loop will be short, this exception to single failure criteria is acceptable.

BECo is investigating alternate methods to satisfy staff requirements. We shall provide NRC with design descriptions and justification as they become available.

SHIFT TECHNICAL ADVISOR (Section 2.2.1.b)

POSITION

Each licensee shall provide an on-shift technical advisor to the shift supervisor. The shift technical advisor may serve more than one unit at a multi-unit site if qualified to perform the advisor function for the various units.

The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have received specific training in the response and analysis of the plant for transients and accidents. The Shift Technical Advisor shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room. The licensee shall assign normal duties to the Shift Technical Advisors that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience.

RESPONSE

The Boston Edison Company will provide a shift technical advisor to the watch engineer on shift at Pilgrim Station. The shift technical advisor will have a bachelor's degree or equivalent in a scientific or engineering dicipline and will receive specific training in the response and analysis of the plant for transients and accidents. A shift technical advisor will be assigned to each operating shift before the beginning of our next operating cycle.

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