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A. V. MORISI
MANAGER
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August 20, 1981
BECo, Ltr, #81-199

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

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

References: A) Letter, USNRC to all Licensees of operating Plants, dated October 1, 1980.

B) Letter, BWR Owners Group position on Isolation on High Radiation dated June 29, 1981.

Dear Sir:

You have requested licensees of operating plants to submit documentation in accordance with the schedule and criteria established in NUREG 0737 "Post TMI Reuirements". Attached you will find out response to the following 0737 requirements.

II.B.1	Reactor Coolant System Vents
II.D.1	Performance Testing of BWR and PWR Relief and Safety Valves
II.E.4.1	Dedicated Hydrogen Penetrations
II.E.4.2(.7)	Containment Isolation Dependability
II.K.3.25	Effect of Loss of A.C. on Pump Seals

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

Very truly yours,

H. R. Balfour for A. V. Morisi

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II.B.1 Reactor Coolant System Vents

References: a) BECo letter #79-206
dated October 19, 1979

b) BWROG letter #8134 D. B. Waters
to D. G. Eisenhut dated
March 31, 1981.

Boston Edison Company has previously addressed the staff's concerns regarding RCS Vents via Reference a), which concluded that adequate reactor coolant system venting is provided by the existing plant design.

In response to the changes to previous requirements and guidance contained in NUREG-0737, the following information is provided:

1) The probability of a valve failing to close, once opened, should be minimized.

RESPONSE

This requirement is interpreted to apply to the ADS safety/relief valves and the RPV continuous vent line valve.

For the ADS safety/relief valves, the probability of a stuck-open relief valve (SORV) is addressed in (Reference b). In this evaluation PNPS-1 would be categorized as a BWR/3 without isolation condenser, with 2-stage Target Rock Valves and low-low set relief or equivalent manual action. Specifically, in Table 5-1 of this evaluation, "SORV Event Frequency" The relative SORV probabality factor of 2-stage Target Rock Valves (P=0.50) has been multiplied by the normalized safety/relief valve challenges for BWR/3 (without isolation condenser) with low-low set relief or equivalent manual action to obtain a SORV event frequency reduction. Using the BWR/3 (without isolation condenser) with 3-stage Target Rock Valves as the bench-mark plant, a reduction in SORV event frequency of more than a factor of ten is observed for plants such as PNPS-1. Thus, the probability of a safety/relief valve failing to close may be judged to be minimal.

When considering the RPV continuous vent line (connecting directly to the main steam piping), a single, manual, normally open 2" globe valve exists which permits venting of non-condensibles with steam to the main turbine. Because this valve is procedurally controlled and verified to be open prior to startup, it is not applicable to address the probability of the valve failing to close, once opened.

2) The reactor coolant vent system should be seismically and environmentally qualified in accordance with IEEE 344-1975...

RESPONSE

Currently, equipment associated with providing reactor coolant venting (i.e., ADS, HPCI/RCIC, RPV vent line) is classified as Seismic Class 1, and is environmentally qualified in accordance with 10CFR50 Appendix A, GDC4. Additional qualification of equipment associated with this system; if deemed necessary, will be addressed via our IE Bulletin 79-01B effort.

3) Provisions to test for operability of the reactor coolant vent system should be part of the design.

RESPONSE

Operability testing is provided in the PNPS-1 Technical Specifications, which require: the ADS safety/relief valves to be tested at pressure during each operating cycle; and simulated automatic actuation tests prior to startup after refueling outage, and in instances when one valve of the ADS is inoperable until the valve is repaired. Additional technical specification requirements for HPCI/RCIC provide operability testing assurance when considering the use of HPCI/RCIC for reactor coolant venting capability.

II.D.1 Performance Testing of BWR and PWR Relief and Safety Valves

References: a) Letter from D. B. Water (BWR Owner's Group) to D. G. Eisenhut (NRC), "Transmittal of Preliminary Data from Generic BWR Safety Relief Valve (s/RV) Test Program" dated July 1, 1981.

b) Letter from D. B. Water (BWR Owner's Group) to R. H. Vollmer (NRC) "Nureg-0578 Requirement 2.1.2-Performance Testing of BWR and PWR Relief.

Reference (a) transmitted the preliminary generic BWR S/RV test program results. Based on our review of this document, we concur that the test program results demonstrate that the tested valve satisfies the acceptance criteria for operability and we hereby confirm that the operational adequacy of the S/RV's for Pilgrim Station has been demonstrated for all test conditions as defined in the test description (Reference (b).

II.E.4.1 Dedicated Hydrogen Penetrations

Dedicated hydrogen recombiner primary containment penetrations were installed during our 1980 refueling outage to facilitate the future installation of hydrogen recombiners if required.

II.E.4.2.(7) Containment Isolation Dependability

Reference: a) BWR Owner's Group letter (T. J. Dente) to U.S. NRC (D. G. Eisenhut) dated June 29, 1981.

Although Boston Edison Company has previously endorsed and committed to the implementation of this NUREG item, this is no longer our position, as we are in full agreement with the BWR Owner's Group evaluation as transmitted to your office via Reference (a) above. This evaluation concludes that, based on existing diverse monitoring capabilities and resultant dose considerations, the automatic closure of containment vent and purge valves on containment radiation is not necessary.

II.K.3.25 Effect of Loss of A.C. on Pump Seals

Reference: a) Letter from D. B. Waters to D. G. Eisenhut, BWROG-8142

Reference a) provided your office with an evaluation on Recirc Pump Seal Leakage which included a description of the seal cooling system, an evaluation of seal performance under loss of offsite A-C power conditions and the results of a seal leakage analysis for extremely deteriorated seals. Pilgrim Station is among those plants for which this evaluation is applicable.

Specifically, at Pilgrim Station, the cooling water to the reactor recirc pump seals is supplied by the RBCCW system. Loop B of the RBCCW systems uses pumps P-202 D, E, and F. Upon loss of offsite power these pumps receive power from the diesel generators (powered from motor control center B14). Thus, pump seal cooling is assured in the event of loss of offsite power. In addition, the seal performance evaluation of Reference (a) concludes that in the unlikely event of a loss of both seals the resultant leakage: 1) does not influence the result of any loss-of-coolant accident analysis; 2) is within the capacity of normal vessel water level control systems and will be easily compensated for; and 3) does not present a safety concern.

Therefore, based on this information, no modifications are planned.