SEP Program

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USNRC-DS

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BECo 87-

U. S. Nuclear Regulatory Commission
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Docket 50-293

ASSESSMENT OF PILGRIM SAFETY ENHANCEMENT PROGRAM

Reference: 1. NRC letter, S.A. Varga to R.G. Bird "Initial Assessment of Pilgrim Safety Enhancement Program," dated August 21, 1987.

2. BECo letter, R.G. Bird to S.A. Varga "Information Regarding Pilgrim Station Safety Enhancement Program," Letter No. 87-111 dated July 8, 1987.

Dear Sir:

The purpose of this letter is to provide additional information in response to the NRC staff's request (Reference 1) regarding the Pilgrim Safety Enhancement Program (SEP), as submitted in Reference 2. The information contained in the attachment to this letter responds to the staff's requests except for those related to the Direct Torus Vent System.

Based on discussions between Mr. J. E. Howard (Boston Edison) and the NRC staff during the period September 23 and 24, 1987, we are deferring our response to the staff's question regarding the Direct Torus Vent System until such time as we can complete additional modeling and analytical work.

If you have any questions or require any additional information, please contact us.

R. G. Bird

Attachment: Assessment of Pilgrim Safety Enhancement Program

MGL/jcp/1159

cc: See next page

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Attachment to BECo Letter No. 87-
Assessment of Pilgrim Safety Enhancement Program

1. Sect. 3.4 - Additional Sources of Water for RPV Injection and Containment Spray

NRC Request

The staff requests clarification regarding the modification to the RHR system to provide additional sources of water for RPV injection and containment spray. This modification may require a change to the Technical Specifications. As described in the enclosure, the valves to be added to the RHR system become part of the reactor coolant pressure boundary during operation of the RHR system and, consequently, are subject to surveillance testing.

BECo Response

No changes are required to the Technical Specifications due to the addition of gate valve 10-HO-511 and check valve 10-CK-510 to the RHR system.

Gate valve 10-HO-511 and check valve 10-CK-510 are not part of the reactor coolant pressure boundary. The reactor coolant pressure boundary consists of all those pressure-retaining components connected to the reactor coolant system, up to and including the outermost containment isolation valve in system piping which penetrates primary reactor containment per 10CFR50.2. Gate valve 10-HO-511 and check valve 10-CK-510 are connected to the RHR system outside the outermost containment isolation valve and are located outside the reactor coolant pressure boundary.

These valves are connected to the RHR system pressure boundary but will be maintained in a closed position during all events or conditions analyzed in the FSAR. The original plant design incorporated a similar connection to the RHR system from the salt service water system. The valves in this interconnection are also maintained in a closed position and are not included in Technical Specifications. The integrity of the RHR system pressure boundary is verified by hydrostatic pressure testing in accordance with the Pilgrim Nuclear Power Station Inservice Inspection Program.

2. Sect. 3.7 - Backup Nitrogen Supply System

NRC Request

The staff requests clarification regarding the function of one valve in the backup nitrogen supply system. As described in the enclosure, valve AO-4356 appears to be a containment isolation valve and, consequently, would be appropriate for inclusion in the Technical Specifications.

BECo Response

Check valve 31-CK-167 is the primary containment isolation valve in the nitrogen supply line, not valve AO-4356 which is upstream of check valve 31-CK-167. Class C lines as defined by FSAR Section 7.3.2 require only one primary containment isolation valve; this is check valve 31-CK-167.

FSAR Table 5.2-5 incorrectly lists AO-4356 as a primary containment isolation valve. The table will be corrected in the next revision of the FSAR. Table 1 of Reference 2.1 also incorrectly lists AO-4356 as a primary containment isolation valve. Therefore, AO-4356 need not be tested in accordance with 10CFR50 Appendix J. AO-4356 is not included in Pilgrim Technical Specifications.

Reference

2.1 BECo letter (J.E. Howard) No. 76-11 to NRC (D.L. Ziemann) "Additional 10CFR50 Appendix J Evaluation", dated January 27, 1976.

3. Sect. 3.12 - Modification to Reactor Core Isolation Cooling System Turbine Exhaust Trip Setpoint

NRC Request

The staff still has questions regarding the proposed modification to the reactor core isolation cooling (RCIC) system. Prior to implementing this modification the staff requests that BECo conduct an assessment of hydrodynamic loads on the RCIC piping and supports, based on the proposed exhaust pressure of 46 psig, and make the results of that assessment available to the staff.

BECo Response

RCIC steam turbine discharge at higher back pressure up to 46 psig is acceptable for the following reasons:

- Starting transients and air clearing loads are low. The RCIC turbine has approximately a 10 second start up time. A gradual start up over such a long time will not produce high air clearing loads or dynamic effects.

- Flow rates through the RCIC exhaust line are low. Steam flow at 25 psig back pressure is 12.3 lbm/ft^2 sec (i.e. 15,350 lbm/hr in an 8" line, Reference FSAR Section 4.7) and would not change appreciably as back pressure is increased to 46 psig.

- The pipe stresses and support/penetration loads for the Reactor Core Isolation Cooling (RCIC) exhaust piping have previously been evaluated for the combined loads of the Mark I Containment Program. The maximum stresses and loads from that program occurred during the simultaneous application of Condensation Oscillation (CO) shell loading, CO drag loads applied to the submerged piping, SSE loads, thermal loads, and weight loads. This load combination controlled the pipe stresses in the Mark I Program because the sinusoidal CO forces occurred in a frequency range where the piping has high dynamic amplification. This is a severe condition that bounds any forces related to continuous steam condensation at the low flow rates associated with RCIC operation.

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- Although pressures and frequencies associated with RCIC discharge cannot be precisely determined, they are bounded, at PNPS, by data from safety relief valve tests. Data recorded in the Monticello Ramshead Tests showed steam condensation loads of ± 5 psid or less. This differential pressure is approximately equal to the cyclic pressures due to the LOCA CO design load, and as discussed above the CO frequencies are in a range of typically high pipe response. The fact that the penetration and supports for the RCIC exhaust piping meet all Code requirements for the CO loads combination clearly demonstrates its ability to withstand the discharge forces associated with its own operation.

- The design pressure of the RCIC turbine discharge piping is 100 psig, which is well above the proposed back pressure setpoint of 46 psig.

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