UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

February 18, 1992

Docket No. 50-293

Mr. Roy A. Anderson
Senior Vice President - Nuclear
Boston Edison Company
Pilgrim Nuclear Power Station
RFD #1 Rocky Hill Road
Plymouth, Massachusetts 02360

Dear Mr. Anderson:

SUBJECT: CLOSURE OF GENERIC LETTER 89-10, SUPPLEMENT 3, "CONSIDERATION OF NRC SPONSORED TESTS OF MOV," (TAC NO. M77793)

On December 17, 1990, and February 26, 1991, the licensee for the Pilgrim Nuclear Power Station submitted a description of its response to Supplement 3 to GL 89-10 for NRC staff review. In a submittal dated August 29, 1991, the licensee answered staff questions on its response to the generic letter supplement. Based on the information provided by the licensee, the NRC staff has not identified any immediate concerns regarding the capability of the MOVs within the scope of Supplement 3 to perform their design basis function to isolate containment in the event of a pipe break downstream of the valves. Therefore, the NRC staff considers it appropriate for the licensee to address the MOVs applicable to Supplement 3 as part of its GL 89-10 program and schedule, unless further information dictates accelerated action.

Among the aspects that the licensee should address are (1) the structural limits of each MOV in light of the increased thrust and torque requirements based on industry experience and research testing, (2) the reduction in thrust delivered by the actuator that may occur as a result of the "rate of loading" phenomenon, (3) the reduction of motor output that may occur as a result of high ambient temperature, (4) the capability of the valves to satisfy any leakage limits associated with the licensee's safety analyses when closing under design basis conditions (particularly where the torque switch is set assuming low valve factors, but is bypassed for a significant portion of the valve stroke), (5) its justification for the assumed stem friction coefficient, (6) its justification for the assumed differential pressure under which the MOVs may be called upon to operate in light of the intent of GL 89-10, (7) the inaccuracy of MOV diagnostic equipment in measuring delivered torque or thrust, (8) the assumed minimum voltage available to the motor as compared to its licensing commitments, and (9) the closing stroke time under design basis conditions in relation to technical specifications or safety analyses (particularly for dc motors). In addition to its own MOV tests, the licensee will be expected to consider MOV tests by other organizations for information on the torque and thrust required to operate the valves under design basis

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conditions. The licensee will be expected to take action to ensure MOV operability where those tests raise questions regarding the required torque or thrust estimates. With respect to the review of the NRC-sponsored MOV tests by the Electric Power Research Institute (EPRI), the NRC staff agrees with the evaluation by the Idaho National Engineering Laboratory (INEL) provided in EGG-SSRE-9926 (November 12, 1991), "Evaluation of EPRI Draft Report NP-9926 - Review of NRC/INEL Gate Valve Test Program."

During inspections of the GL 89-10 program, the NRC staff will confirm the licensee's assumptions and calculations for MOVs within the scope of Supplement 3 as well as the other MOVs within the scope of GL 89-10. This letter closes TAC NO. M77793.

Sincerely,

Original signed by:

Ronald B. Eaton, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

cc:
See next page

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February 20, 1992

Docket No. 50-029

NOTE TO: File

FROM: Daniel H. Dorman, Project Manager
Yankee Rowe Review Group
Division of Reactor Projects - I/II

SUBJECT: INFORMATION REGARDING THE YANKEE ROWE REACTOR PRESSURE VESSEL (RPV)

On January 30, 1992, the staff received a document, "The Effect of Post-Irradiation Annealing and Re-Irradiation on the Fracture Properties of A302B Pressure Vessel Steel," dated July 1973, from the Yankee Atomic Electric Company (YAEC), which was made available to the public in a note to file dated January 31, 1992. That copy was missing pages 12 and 13. Those pages were provided by YAEC on February 19, 1992 and are provided in Enclosure 1.

Enclosure 2 is a report on Commissioner Rogers' visit to Southwest Research Institute on January 27-28, 1992, from Jack C. Scarborough, dated February 11, 1992.

original signed by:
Daniel H. Dorman, Project Manager
Yankee Rowe Review Group
Division of Reactor Projects - I/II

Enclosures:
As stated

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DATE	02/20/92	02/20/92	02/20/92		

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TABLE I

HEAT TREATMENTS APPLIED TO TEST MATERIALS

Material S

Austenitized at 1650-1700°F for 4 hours, air cooled, tempered at 1200-1250°F, stress relieved at 1150°F for 4 hours (up to 6 cycles).

Material U

Austenitized 1650-1700°F for 4 hours, water quenched to 1000°F, air cooled, tempered at 1200-1250°F, stress relieved at 1150°F for 4 hours (up to 6 cycles).

Material P

Austenitized 1650-1675°F for 6 hours, water spray quenched to 500°F, tempered at 1150-1200°F for 4 hours, furnace cooled 50°F/hours to 600°F, air cooled, stress relieved 1125-1150°F for 24 hours, furnace cooled 50°F/hours to 600°F, air cooled.

CHEMICAL COMPOSITION OF TEST MATERIALS

Material	C %	Mn %	Si %	S %	P %	Mo %	Ni %	Cr %	Al %	Co %	O PPM	Cu
S Mill Analysis	.21	1.40	.22	.035	.027	.46	-	-	-	-	-	-
Bettis Check	.22	1.53	.47	.031	.023	.47	.08	.08	.028	.14	142	.14
U Mill Analysis	.19	1.17	.21	.027	.015	.48	-	-	-	-	-	-
Bettis Check	.21	1.34	.22	.027	.021	.50	.13	.14	.045	.22	169	.22
P Mill Analysis	.19	1.50	.27	.026	.012	.48	-	-	-	-	-	-
Bettis Check	.22	1.74	.31	.017	.008	.46	.12	.10	.013	.20	158	.20