December 5, 2000

Mr. Mike Bellamy
Site Vice President
Entergy Nuclear Generation Company
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, MA 02360

SUBJECT: COMPLETION OF LICENSING ACTIVITY FOR GENERIC LETTER 96-06, "ASSURANCE OF EQUIPMENT OPERABILITY AND CONTAINMENT INTEGRITY DURING DESIGN BASIS ACCIDENT CONDITIONS," PILGRIM NUCLEAR POWER STATION (TAC NO. M96851)

Dear Mr. Bellamy:

Generic Letter (GL) 96-06, “Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions,” was issued on September 30, 1996. The GL requested licensees to determine (1) if containment air cooler cooling water systems are susceptible to either water-hammer or two-phase flow conditions during postulated accident conditions, and (2) if piping systems that penetrate the containment are susceptible to thermal expansion of fluid so that overpressurization of piping could occur. The 120-day response to GL 96-06 for Pilgrim Nuclear Power Station was submitted by Boston Edison Company (now Entergy Nuclear Generation Company/the licensee) to the Nuclear Regulatory Commission (NRC) by letter dated January 28, 1997. The NRC staff reviewed the response and requested additional information in a letter dated July 14, 1998. In a letter dated October 1, 1998, the licensee provided the additional information. By letter dated November 30, 1998, the licensee provided a response to a second request for additional information regarding GL 96-06 forwarded by the NRC staff on October 29, 1998. The licensee provided updated GL 96-06 responses by letters dated May 24 and July 29, 1999.

Water-Hammer or Two-Phase Flow

Cooling water systems serving the containment air coolers may be exposed to the hydrodynamic effects of water-hammer during either a loss-of-coolant accident or a main steamline break. In addition, cooling water systems serving the containment air coolers may experience two-phase flow conditions during design-basis accident scenarios whereas the heat removal assumptions were based on single-phase flow conditions. Therefore, cooling water systems may need corrective actions to satisfy system design and operability requirements.

The licensee has stated that: (a) the water-hammer and two-phase flow concerns are only applicable to Loop B of the reactor building closed cooling water (RBCCW) system, and that Loop A is not affected because only Loop B supplies cooling water to the drywell coolers; (b) the heat removal function of the containment air coolers (CACs) is not required for accident mitigation purposes; and (c) water-hammer could occur in the CACs as a consequence of a single active failure associated with RBCCW, Loop B, but the applicable containment isolation

valves would not be affected and could subsequently be isolated by the operators to eliminate the possibility of containment bypass leakage. In addition, as an added precaution to preclude water-hammer, Procedure 2.2.19.5, “RHR Modes of Operation for Transients,” and Procedure 2.4.42, “Loss of RBCCW,” have been revised to prevent operators from initiating RBCCW flow through the CACs when the drywell temperature exceeds 250 °F.

Based on the information that was submitted (as described above), the staff agrees that since the CACs are not safety-related, the consequence of the RBCCW Loop B line failure is the loss of containment integrity. The operators, in this case, will be able to close the containment isolation valves associated with the failed piping in the RBCCW system, Loop B, to eliminate any containment bypass leakage. Loop A of the RBCCW system is not affected by this waterhammer event. We are satisfied with the licensee’s response and consider the water-hammer and two-phase flow elements of GL 96-06 to be closed. Our conclusion is based on our review of your deterministic evaluation that assumes a single active failure that leads to a water-hammer-induced piping failure in Loop B of the RBCCW system.

Therefore, the staff concludes that your response adequately addresses the water-hammer and two-phase flow concerns identified in the GL.

Piping Overpressurization

Thermally induced overpressurization of isolated water-filled piping sections in containment could jeopardize the ability of accident-mitigating systems to perform their safety functions and could also lead to a breach of containment integrity via bypass leakage. Corrective actions may be needed to satisfy system operability requirements.

In its submittal of January 28, 1997, the licensee identified six pipe penetrations potentially vulnerable to a water solid volume that may be subjected to an increase in pressure due to heating of the trapped fluid. Entergy has performed the following modifications or analysis to address overpressurization in the six pipe penetrations:

1) The licensee indicated that, for the two RBCCW lines, protection would be provided by an existing relief valve. The licensee has replaced this relief valve during refueling outage (RFO) 11. In addition, this relief valve was added to the inservice testing program.

2) The licensee has added two pressure relief devices to two sump pump discharge lines.

3) The licensee has opened the valve inside containment on the core spray sample line to eliminate an isolated section of the line.

4) With regard to the residual heat removal (RHR) shutdown cooling (SDC) suction line, the licensee provided analysis that demonstrated the pressure in this line remains below its maximum operating pressure during postulated design-basis loss-of-coolant accident and main steam line break events.

On October 29, 1998, the staff requested additional information regarding the analysis performed with regard to the RHR SDC suction line. In its response dated November 30, 1998,

the licensee provided the applicable design criteria and methodology for calculating maximum temperature and pressure in the RHR SDC suction line. In Inspection Report 50-293/98-203, the staff identified potentially non-conservative assumptions in the calculations performed in response to GL 96-06. By letter dated July 29, 1999, the licensee responded to this staff concern. The licensee performed a new analysis of the RHR SDC suction line based on an ANSYS finite element thermal model and a higher value of the heat transfer film coefficient applicable to the internal convection conditions for the isolated fluid to determine the pressure in the line. The licensee applied the calculated pressure to the ANSYS finite element structural model of the RHR SDC suction line and evaluated the pipe stress due to internal pressure, dead weight, and safe shutdown earthquake loads. The licensee determined the combined pipe stress to be within the Final Safety Analysis Report design-basis pipe stress limit. The staff finds this evaluation to be acceptable.

Based on this information, the staff concludes that your response adequately addresses the piping overpressurization concerns identified in the GL.

This action closes the staff's activities associated with GL 96-06. If you have any questions, please call me at (301)415-1445.

Sincerely,

/RA/

Alan B. Wang, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-293

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