50-293

Mr. E. Thomas Boulette, Ph.D
Senior Vice President - Nuclear
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Pilgrim Nuclear Power Station
RFD #1 Rocky Hill Road
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February 25, 1997

SUBJECT: REVIEW OF 10 CFR 50.59 EVALUATION REGARDING NET POSITIVE SUCTION HEAD (NPSH) CALCULATION FOR THE PILGRIM NUCLEAR POWER STATION (TAC NO. M96176)

Dear Mr. Boulette:

The NRC recently began an audit of your 10 CFR 50.59 analysis in support of a change to the maximum allowable service water temperature. Region I requested NRR support of this audit. An internal report was produced by the Reactor Systems Branch (SRXB) for the Containment System Branch (SCSB) in support of this audit. The SRXB input was inadvertently placed in the Public Document Room (PDR). Because the document was placed in the PDR, the staff is providing Boston Edison Company (BECo) with the SRXB preliminary input as it is available to any interested party via the PDR. Although SRXB has questioned the acceptability of BECO's conclusions regarding an unreviewed safety question, the NRC has not made any final conclusions as the SRXB input must be integrated with input from the SCSB, before the NRC makes any final conclusions.

Should questions exist regarding this matter, please contact Mr. Alan Wang at (301) 415-1445.

Sincerely,

(Original Signed By)

Patrick D. Milano, Acting Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure: Safety Evaluation

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NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 25, 1997

Mr. E. Thomas Boulette, Ph.D
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SUBJECT: AUDIT OF NET POSITIVE SUCTION HEAD (NPSH) 10 CFR 50.59 ANALYSIS FOR THE PILGRIM NUCLEAR POWER STATION (TAC NO. M96176)

Dear Mr. Boulette:

The NRC recently began an audit of your 10 CFR 50.59 analysis in support of a change to the maximum allowable service water temperature. Region I requested NRR support of this audit. An internal report was produced by the Reactor Systems Branch (SRXB) for the Containment System Branch (SCSB) in support of this audit. The SRXB input was inadvertently placed in the Public Document Room (PDR). Because the document was placed in the PDR, the staff is providing Boston Edison Company (BECo) with the SRXB preliminary input as it is available to any interested party via the PDR. Although SRXB has questioned the acceptability of BECO's conclusions regarding an unreviewed safety question, the NRC has not made any final conclusions as the SRXB input must be integrated with input from the SCSB, before the NRC makes any final conclusions.

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Docket No. 50-293

Enclosure: Audit

cc w/encl: See next page

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AUDIT REVIEW REGARDING THE PILGRIM LICENSING BASIS AND SUBSEQUENT MODIFICATION UNDER 10 CFR 50.59 FOR TWO SAFETY SIGNIFICANT PROBLEMS

1.0 INTRODUCTION

By letter dated July 18, 1996, Region I requested assistance to determine the acceptability of the Boston Edison Company (BECo) conclusion that no Unreviewed Safety Questions (USQ) exist for two safety significant problems at the Pilgrim Nuclear Power Station (PNPS). The questions posed to the staff are as follows:

1. "Can BECo credit post-accident containment pressure to offset the pressure drop caused by debris laden emergency core cooling system (ECCS) suction strainers without NRC staff review and approval in accordance with the 10 CFR 50.59 process; and, overall, is the BECo safety evaluation for debris clogging of ECCS suction strainers adequate?"

2. "Can BECo rely on the 'SHEX' computer code for modeling Pilgrim post-accident containment temperature and pressure as a result of operating the service water system with inlet temperature up to 75 degrees F without NRC staff review and approval in accordance with the 10 CFR 50.59 process; and, overall, is the BECo safety evaluation for inlet temperatures from 65 to 75 degrees F adequate with respect (to) the licensing and design basis for the plant?"

The Reactor Systems Branch (SRXB) has addressed part one of question one regarding whether BECo can credit post-accident containment pressure with staff review and approval. Our review is as follows.

2.0 BACKGROUND

The Pilgrim Safety Evaluation Report (SER) by the AEC (Reference 1), dated August 25, 1971, discussed net positive suction head to RHR and Core Spray pumps in Section 5.7.

5.7 Net Positive Suction Head (NPSH) to RHR and Core Spray Pumps

During the course of the construction permit review on Pilgrim, we questioned whether the RHR and core spray pumps, and their respective systems, were designed to provide an adequate NPSH margin to assure their continued operation following a loss-of-coolant accident. In Amendment 9 to the application, Boston Edison Company furnished an analysis based on preliminary design assumptions showing that a positive NPSH margin would be available following the accident without requiring containment overpressure. The applicant provided further information in Amendment No. 24 with an analysis confirming the final design requirement that a positive NPSH margin be available even if the containment spray were operating following a design basis loss-of-coolant accident (LOCA). We conclude that the equipment provided is adequate to assure sufficient NPSH to the emergency system pumps in the unlikely event of a LOCA.

Enclosure

In order to understand what may have been the thought processes of the AEC staff at the time the AEC SER was written, the staff reviewed Amendments 9 and 24, Safety Guide 1, and other supporting documentation. In Amendment 9 to the Preliminary Safety Analysis Report (PSAR) (Reference 2), the licensee responded to the following AEC question:

> "Describe and justify the NPSH requirements for the RHRS and Reactor Core Spray Cooling System Pumps."

As stated in the AEC SER, Amendment 9 provided a preliminary design analysis of the total NPSH available for Pilgrim. The conservative model used to calculate NPSH available assumed atmospheric pressure, 0 psig, and 100% humidity in both the drywell and wetwell, with respective temperatures of 150 °F and 80 °F. At that time, 28 feet was the NPSH required for the RHR and the core spray pumps. Figure 1-2 of the submittal showed the containment pressure required for 28 feet of NPSH during a design basis LOCA. From Figure 1-2 it can be seen that the peak containment pressure required was approximately 13.8 psia. This value is below 1 atmosphere, and hence, the difference between 1 atmosphere and the peak containment pressure required for 28 feet of NPSH is positive margin. Thus, the staff concluded that no containment overpressure was required and positive margin existed.

In addition, the following AEC question was addressed in Amendment 24 (Reference 3):

> "Provide data on containment pressure (with spray) vs time following a LOCA showing required pressure to assure adequate NPSH to core cooling pumps. Consider data and write up given in Quad Cities Amendments 16 and 17. These data relate to various levels of system degradation."

The licensee's response to the question was as follows:

> "A complete analysis of total NPSH available has been previously documented in Pilgrim Amendment 9, Comment 1. This analysis remains conservative with respect to the model and assumptions utilized. The attached Figure which more closely corresponds to final station design confirms the fact that substantial NPSH margin will be available at all times following a design basis loss-of-coolant accident, both with and without containment spray."

The licensee did not address the Quad Cities amendments but provided a figure which shows that peak containment pressure required is below one atmosphere during the design basis LOCA with containment spray. The staff notes that no other data was provided in Amendment 24 besides the figure. The staff reviewed the Quad Cities amendments (References 4 and 5) which showed that Quad Cities needed approximately 8 hours of containment overpressure to assure adequate NPSH to the RHR pumps in LPCI mode following a design basis LOCA. Several differences were noticed between Pilgrim and Quad Cities. In Pilgrim's NPSH calculations, the initial suppression pool temperature was assumed to 80 °F whereas Quad Cities was assumed to be 90 °F; Pilgrim's core spray pumps have a higher NPSH requirement, 28 ft., than the RHR pumps,

23 ft., however at Quad Cities, the RHR pumps in LPCI modes had the higher NPSH requirements of the two pumps. Also, the Quad Cities RHR service water pumps are required to operate during cooldown of the reactor vessel during operation. This required the Quad Cities pumps to be designed with higher head capability. Based on this information, it appears that the AEC was looking for information regarding the changes in RHR flow requirements with and without containment spray and its effect on RHR NPSH requirements. Since Amendment 24 did not show that containment overpressure was needed for adequate NPSH of both pumps with containment spray, it was concluded by the staff that positive margin was available. As stated above, this positive margin is the difference between one atmosphere and the peak containment pressure required for adequate NPSH during LOCA conditions.

3.0 EVALUATION

BECo developed Safety Evaluation 2971 (Reference 6) to evaluate the effects of fiberglass insulation debris on Pilgrim emergency core cooling system (ECCS) performance (a.k.a. Core Standby Cooling Systems (CSCS)). SE 2971 references Calculation M-662, "RHR and Core Spray Pump NPSH and Suction Pressure Drop," and GE Report GE-NE-B13-01805-11, "Effects of Fiberglass Insulation Debris on Pilgrim ECCS Pump Performance," (References 7 and 8). Calculation M-662 provides an analysis of NPSH conditions for RHR and Core Spray Pumps during performance test conditions and following the design basis loss of coolant accident. The staff notes that data generated in this calculation is currently used for the applicable figures, i.e., 14-5.9, 10, and 13, in Chapter 14 of the FSAR (Reference 9), and are based on clean strainer conditions.

The current model, as described in SE 2971, used to calculate NPSH available assumes initial conditions of 1.3 psig, 150 °F, and 80% humidity in the drywell and 0 psig, 80 °F, and 100% humidity in the wetwell. The licensee stated that the original initial drywell conditions were not credible conditions to exist prior to a postulated accident. The licensee continued to state that although the original analysis appears more conservative in this regard, there was no basis for the inconsistent assumptions on the dry well conditions. It is unclear to the staff as to how the drywell assumptions are inconsistent. The licensee used the new NPSH available model to verify NPSH available with strainers laden with debris. The licensee's Calculation M-622 shows that NPSH available is greater than NPSH required with debris laden strainers. However, the staff has performed verification calculations using the licensee's original and new initial conditions, new suction losses, new NPSH required, currently 29 ft., and strainer head losses due to debris. At worst case conditions, the staff verified that NPSH available was greater than NPSH required using the new initial conditions. However, NPSH available was not greater than NPSH required during worst-case conditions using the original initial conditions. Worst-case conditions are torus temperatures of 166 °F and 178 °F which correspond to service water inlet temperatures of 65 °F and 75 °F, respectively.

In order to clarify their licensing basis with regard to positive containment pressure, the licensee discussed the curve submitted in Amendment 24 which showed the effect of continuous containment spray on available containment pressure and NPSH available for a DBA LOCA. The licensee states that "as shown for the spray case, the containment pressure is at or above atmospheric pressure while the pressure required to meet NPSH requirements is always less than atmospheric pressure, hence NPSH available exceeds NPSH required during continuous spray for the nominal design case. The original FSAR Figure 14.5-13 does not reflect the drywell spray reduction implemented via PDC86-52A or the potential effects of fibrous debris on available NPSH." The staff notes that the new FSAR Figure 14.5-13 does not resemble the original FSAR Figure 14.5-13 which was submitted in Amendment 24. The current Figure 14.5-13 also takes credit for positive containment pressure as stated above. The staff does not agree with the licensee.

In the AEC SER, the staff stated that containment overpressure was not required to have positive NPSH margin during DBA LOCA conditions. This is a statement of fact, not a requirement. Although Safety Guide 1 was issued by the time the AEC SER was issued, Safety Guide 1 is not part of the Pilgrim licensing basis. With a service water inlet temperature of 65 °F and clean ECCS strainers, no containment overpressure required for positive NPSH margin (margin below one atmosphere) remains a statement of fact. However, due to the new insulation debris, changes to the assumptions of the initial conditions, and higher service water inlet temperature, discussed in another SE, the staff believes that probability of a malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased. Since debris from insulation blown off of pipes during a DBA LOCA increases the probability of malfunction in the ECCS pumps resulting from cavitation, the staff believes a USQ exists as described in 10 CFR 50.59(a)(2)(i). As such, changes to the insulation which can cause debris on the ECCS suction strainers which have limited margin for NPSH available should have been submitted to the staff for review and approval.

4.0 CONCLUSION

Based on the above information, the staff believes a USQ exists regarding the installation of insulation on pipes which can cause debris. Also, the staff does not concur with the licensee that Pilgrim's licensing basis included credit for positive containment pressure. As stated earlier, Safety Guide 1 is not part of Pilgrim's licensing basis; however, neither is containment overpressure.

5.0 REFERENCES

1. "Safety Evaluation by the Division of Reactor Licensing U.S. Atomic Energy Commission in the Matter of Boston Edison Company Pilgrim Nuclear Power Station," August 25, 1971.

2. Boston Edison Company, Pilgrim Nuclear Power Station, Amendment 9, March 11, 1968.

3. Boston Edison Company, Pilgrim Nuclear Power Station, Amendment 24, March 18, 1971.

4. Byron Lee, Jr., Commonwealth Edison Company, to Dr. P. A. Morris, USAEC, "Amendment 16 to the Applications for Construction Permits and Operation Licenses for Quad Cities Units 1 and 2," February 8, 1971.

5. Byron Lee, Jr., Commonwealth Edison Company, to Dr. P. A. Morris, USAEC, "Amendment 17 to the Applications for Construction Permits and Operation Licenses for Quad Cities Units 1 and 2," March 1, 1971.

6. Safety Evaluation 2971, Pilgrim Nuclear Power Station, March 25, 1996.

7. Calculation M-662, Rev. E1, "RHR and Core Spray Pump NPSH and Suction Pressure Drop," March 20, 1996.

8. GE Document, GE-NE-B13-01805-11, "Effects of Fiberglass Insulation Debris on Pilgrim ECCS Pump Performance," January 1996.

9. Pilgrim Nuclear Power Station Final Safety Analysis Report, Revision 19, June 1996.