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

July 3, 1997

Mr. E. Thomas Boulette, Ph.D
Senior Vice President - Nuclear
Boston Edison Company
Pilgrim Nuclear Power Station
RFD #1 Rocky Hill Road
Plymouth, MA 02360

SUBJECT: ISSUANCE OF AMENDMENT NO. 173 TO FACILITY OPERATING LICENSE NO. DPR-35, PILGRIM NUCLEAR POWER STATION (TAC NO. M97789)

Dear Mr. Boulette:

The Commission has issued the enclosed Amendment No. 173 to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station. This amendment is in response to your application dated January 20, 1997, as supplemented January 30, February 27, April 11, May 14, and June 20 (2 letters), 1997.

The proposed amendment would (1) evaluate the unreviewed safety questions (USQ) associated with the operation of Pilgrim Nuclear Power Station to credit the use of containment overpressure to supplement the net positive suction head (NPSH) for the emergency core cooling pumps and increase the accident analysis design ultimate heat sink (UHS) temperature from 65 °F to 75 °F, (2) change the UHS administrative limit from 68 °F to 75 °F, and (3) authorize the licensee to change the Updated Final Safety Analysis Report (UFSAR) to reflect the use of containment pressure to compensate for the deficiency in NPSH following a design basis accident and increase the accident analysis design UHS temperature from 65° F to 75° F. Boston Edison Company (BECo) has proposed to submit a Technical Specification amendment for the UHS temperature by the first quarter of 1998. In addition, within 180 days of issuance of this amendment, BECo has committed to complete the containment analysis using the ANS 5.1-1979 Decay Heat Curve with a 2-sigma uncertainty added. The NRC staff finds this amendment change is acceptable as specified in the enclosed Safety Evaluation. During this review the staff identified two potential exceptions from 10 CFR 50.49 and will resolve these issues independently from this amendment.

cc:

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A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register Notice.

Sincerely,

Original signed by:

Alan B. Wang, Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-293

Enclosures: 1. Amendment No. 173 to License No. DPR-35
2. Safety Evaluation

cc w/encls: See next page

subject to changes noted

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AMENDMENT NO. 173 TO FACILITY OPERATING LICENSE NO. DPR-35-PILGRIM NUCLEAR POWER STATION

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

BOSTON EDISON COMPANY

DOCKET NO. 50-293

PILGRIM NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 173
License No. DPR-35

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The application for amendment filed by the Boston Edison Company (the licensee) dated January 20, 1997, as supplemented January 30, February 27, April 11, May 14, and June 20 (2 letters), 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended to authorize changes to the description of the facility in the Updated Final Analysis Report (UFSAR) as set forth in the application for amendment by Boston Edison Company dated January 20, 1997, as supplemented January 30, February 27, April 11, May 14, and June 20 (2 letters), 1997.

   In addition, the license is amended to include the following conditions, which shall be added to BECo's "Index of Technical Specification Changes."

   a. Update the UFSAR to credit containment pressure in the design basis accident including the 75 °F seawater temperature design change as part of the next UFSAR update,

b. Change the administrative limit from 68 °F to 75 °F for entering the loss of containment cooling limiting condition of operation and propose an ultimate heat sink Technical Specification by the first quarter of 1998, and

c. Within 180 days of issuance of the amendment complete the containment analysis using the ANS 5.1-1979 Decay Heat Curve with a 2-sigma uncertainty added.

3. This license amendment is effective as of its date of issuance and shall be implemented within 180 days from the date of issuance. Implementation of the amendment is the incorporation in the UFSAR of the changes to the description of the facility as described in the licensee's application dated January 20, 1997, as supplemented January 30, February 27, April 11, May 14, and June 20 (2 letters), 1997, and evaluated in the Staff's Evaluation attached to this amendment. Implementation also includes fulfillment of the specified conditions set forth in paragraph 2, except that the Licensee shall have up to the first quarter of 1998 to submit an ultimate heat sink Technical Specification.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Date of Issuance: July 3, 1997

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 173 TO FACILITY OPERATING LICENSE NO. DPR-35

BOSTON EDISON COMPANY

PILGRIM NUCLEAR POWER STATION

DOCKET NO. 50-293

1.0 INTRODUCTION

The proposed amendment would (1) evaluate the unreviewed safety questions (USQ) associated with the operation of Pilgrim Nuclear Power Station to credit the use of containment overpressure to supplement the net positive suction head (NPSH) for the emergency core cooling pumps and increase the accident analysis design ultimate heat sink (UHS) temperature from 65 °F to 75 °F, (2) change the UHS administrative limit from 68 °F to 75 °F, and (3) authorize the licensee to change the Updated Final Safety Analysis Report (UFSAR) to reflect the use of containment pressure to compensate for the deficiency in NPSH following a design basis accident and increase the accident analysis design UHS temperature from 65° F to 75° F. As part of this amendment, Boston Edison Company (BECo/licensee) has proposed to submit a Technical Specification amendment for the UHS temperature by the first quarter of 1998. In addition, within 180 days of issuance of this amendment, BECo has committed to complete the containment analysis using the ANS 5.1-1979 Decay Heat Curve with a 2-sigma uncertainty added.

In 1984 BECo replaced recirculation system piping at the Pilgrim Nuclear Power Station (PNPS) and, along with the modification, also replaced the drywell reflective metal insulation with NUKON insulation. Safety Evaluation #1638 was prepared to evaluate this modification under the 10 CFR 50.59 process. This plant modification resulted in the need to credit containment pressure for sufficient net positive suction head (NPSH) for emergency core cooling system (ECCS) pumps following a loss-of-coolant accident (LOCA). However, this credit for containment pressure was not recognized at the time of the review. In 1995 while performing additional analysis to justify and increase service water temperature, this error was discovered in Safety Evaluation #1638. Safety Evaluation #2971 was prepared to supersede #1638 to credit containment overpressure. Safety Evaluation #2983 was prepared to increase the service water temperature from 65 °F to 75 °F. BECo concluded that credit for the use of containment pressure and the change in service water temperature are not unreviewed safety questions (USQ) and could be made without staff review and approval.

In July 1996, Region I requested assistance in evaluating the acceptability of the BECo conclusion that no USQ exists for the credit for use of containment pressure and the change in service water temperature at the PNPS. By letter dated February 12, 1997 (Reference 8), the NRR staff determined that credit for post-accident containment overpressure to offset the pressure drop caused by debris laden ECCS suction strainers is a USQ. As a result of the staff's review, BECo by letter dated January 20, 1997 (Reference 1) with the proposed no significant hazards consideration submitted by letter on January 30, 1997, as supplemented by letters dated February 27, April 11, May 14, and June 20, 1997 (2 letters) (References 2, 3, 4, 5, 6, and 7), BECo requested changes to the PNPS licensing basis as described in the Updated Final Safety Analysis Report (UFSAR). During the last refueling outage (RFO #11) in winter 1997, the licensee replaced their existing ECCS strainers with larger strainers. Based on this activity, the licensee has requested review and approval of credit for a limited amount of containment overpressure to compensate for a slight increase in the NPSH deficiency post design basis accident. The February 27, April 11, May 14, and June 20 (2 letters), 1997, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

In addition the licensee has also proposed to change the licensing basis service water inlet temperature from 65 °F to 75 °F. The ultimate heat sink (UHS) at the PNPS is the Atlantic Ocean. Sea water is taken from Cape Cod Bay by the salt (station) service water (SSW) system pumps which supply cooling water to the heat exchangers of the reactor building closed cooling water (RBCCW) system and the turbine building closed cooling water (TBCCW) system. The safety-related RBCCW is required for safe plant shutdown and functions during normal operation and following design basis accidents. The non-safety-related TBCCW system is not necessary for safe plant shutdown and is not required to operate during or following any design basis accidents. Because the SSW system supplies cooling water to the RBCCW system it is also safety related and operates during both normal and accident conditions. The UFSAR used a SSW injection temperature (inlet to the RBCCW heat exchanger) of 65 °F (degrees Fahrenheit) for all accident and transient analyses (30 days for the design basis LOCA). Licensee review of recent operating history determined that the cyclic temperatures in the bay due to ocean currents combined with unusually high outdoor area temperatures have resulted in conditions where the SSW injection temperature has momentarily exceeded the 65 °F analyzed temperature. As a result of the potential for exceeding the analyzed conditions due to increasing trends in the SSW injection temperature, the licensee has reanalyzed the design basis events (with the design basis LOCA being most limiting) assuming a constant SSW injection temperature of 75 °F.

The January 20, 1997, submittal included a licensee safety evaluation (SE #2983) of proposed revisions to the UFSAR to reflect the new analysis results. In Attachment A to SE #2983, the licensee identified that measurements in excess of 75 °F are expected to occur in 1 year out of 20; and in that year, the temperature may exceed 75 °F on more than one occasion but not for extended periods of time. However, the licensee has only provided analysis for, and the staff's evaluation only addresses a SSW injection temperature of up to and including 75 °F. Therefore, operation above 75 °F would be considered outside the design and licensing basis for the plant and, if exceeded, would require further justification by the licensee. To address this potential concern the licensee in its May 14, 1997, submittal committed to submit an UHS temperature TS limit by the end of the first quarter of 1998.

2.0 DISCUSSION

The SSW supplies cooling water to the heat exchangers of the RBCCW system and the TBCCW system. This SE addresses the affects of SSW system temperature increase on the safety-related components cooled by the RBCCW system. The staff has determined that components cooled by the TBCCW system are not safety related and are not necessary for safe plant shutdown. Therefore, the increase in TBCCW system temperature does not affect safe plant shutdown and is acceptable based on that determination. The safety functions affected by the increase in RBCCW temperature (due to increased SSW temperature) are emergency core cooling, containment cooling, component or equipment cooling, and compartment or area (ventilation) cooling.

The primary effect of an increased SSW injection temperature is an increase in the RBCCW temperature and corresponding temperature increase in the suppression pool, and other areas of the plant influenced by the RBCCW temperature. The RBCCW system provides a heat sink for the residual heat removal (RHR) heat exchangers and provides cooling to the core standby cooling systems (CSCS). The CSCS are comprised of the core spray (CS) system, low-pressure core injection (LPCI) system, and the high-pressure core injection (HPCI) system. Specific CSCS equipment cooled by the system are the RHR (used for LPCI) pump seal coolers, CS motor coolers and area coolers serving the CSCS equipment. The increase in RHR system temperature (via the RHR heat exchangers) affects suppression pool cooling, LPCI with heat rejection, and containment spray. Therefore, containment cooling in addition to core cooling is affected by this temperature increase. The higher containment temperature and pressure in turn, affect the equipment qualification (EQ) profiles used to qualify equipment necessary to function following a design-basis LOCA. Also, the increase in suppression pool temperature affects the NPSH available to the CSCS pumps. The following evaluation addresses the need for containment pressure for the NPSH of the ECCS pumps due to the temperature increase of the UHS and the use of fibrous insulation on the recirculation piping.

3.0 EVALUATION

3.1 Minimum Containment Pressure/Maximum Suppression Pool Temperature Analysis

3.1.1 Methodology and Assumptions

To calculate the peak suppression pool temperature for the limiting NPSH case (a double-ended break of a recirculation suction line), the licensee had General Electric conduct analyses (Reference 13) with the SHEX computer code for the first 24 hours following the design-basis LOCA.

Beyond 24 hours, a simpler model within the SHEX code was used to calculate the suppression pool temperature response. The model performs a mass and energy balance between the drywell and wetwell atmospheres, and provides the input for BECo Calculation M-662, Rev. E2 (Reference 10), which is the design-basis calculation for available NPSH. In this calculation, the suppression pool temperature is used to calculate the minimum containment pressure by using an equilibrium formulation to calculate the drywell airspace pressure.

This formulation assumes that the drywell steam space is in equilibrium with the suppression pool vapor pressure, such that these two terms cancel in the equation for the available NPSH. The drywell airspace is assumed to pressurize as an ideal gas at the same temperature as the suppression pool. The same methodology was used in the original NPSH calculation as described in the UFSAR.

To calculate conservative NPSH available values, the initial conditions and modeling assumptions used in the SHEX calculation were chosen to maximize the peak suppression pool temperature, and the methodology and input assumptions used to calculate the containment pressure in Calculation M-662 were chosen to minimize the calculated containment pressure. These assumptions include the following:

1) Initial thermal power of 102 percent rated thermal power.

2) Initial suppression pool temperature of 80 °F.

3) Continued feedwater addition to the reactor following the LOCA.

4) TS minimum torus water level.

5) Rated horsepower (converted to heat) from ECCS pumps is added to ECCS flow.

6) Minimum RHR heat exchanger flow rates and maximum allowable fouling and tube plugging.

7) 75 °F service water inlet temperature.

8) ANS 5.1-1979 decay heat, with and without a 2-sigma uncertainty added.

9) Initiation of lower-than-rated containment cooling at 600 seconds, switching to rated containment cooling at 2 hours from beginning of LOCA.

10) Initial drywell pressure of 1.3 psig, and initial wetwell pressure of 0.0 psig.

11) Initial relative humidity of 100 percent in the drywell, and 80 percent in the wetwell.

12) Initial temperature of 150 °F in the drywell, and 80 °F in the wetwell.

13) The effects of containment leakage on the calculated pressure were considered.

The first seven of these assumptions tend to maximize the peak calculated suppression pool temperature, while the last four tend to minimize the containment pressure.

In a request for additional information (RAI) dated March 13, 1997, the staff asked the licensee why it believed the use of an ideal gas/equilibrium formulation to calculate the containment pressure was adequate, and to discuss the conservatism of the method used. In its response (Reference 5), the licensee indicated that the equilibrium method used in the present analysis is consistent with the method used in the original design basis calculations. Furthermore, the licensee indicated that a mechanistic analysis (i.e., conducted with SHEX) for the case of a steamline break with containment sprays operating is less limiting in terms of NPSH margin than the double-ended recirculation line break using the equilibrium methodology described above, based on a comparison of the equilibrium methodology to calculations conducted by General Electric (References 13 and 5).

In response to an RAI (Reference 5), the licensee further clarified the methodology used to calculate the minimum containment pressure, and clarified why it believed the initial conditions chosen for the minimum pressure analysis are conservative. In particular, to use the ideal gas law to calculate the containment pressure, the initial noncondensable gas mass was calculated assuming the aforementioned initial pressures, temperatures, and relative humidities to yield the gas mass of the mixed drywell and wetwell volumes. While the staff would typically expect that a relative humidity of 100 percent be used in the drywell, the licensee indicated that a humidity of 100 percent at 150 °F would cause the pressure to exceed the scram setpoint for containment pressure. The licensee used what it believed to be a more appropriate value of 80 percent, while still maintaining an adequate degree of conservatism.

The staff asked the licensee to provide additional justification that the assumed initial temperature of the drywell was conservative. The licensee submitted an average temperature calculation (Reference 7) derived from instruments located at various elevations in the drywell. The licensee's calculation indicates a maximum normal temperature of 135 °F at 68 percent relative humidity with a drywell pressure of 1.6 psig. Comparing this temperature with the assumed temperature of 150 °F at 80 percent relative humidity and 1.3 psig, the staff notes that the higher temperature and humidity of 150 °F and 80 percent used in the analysis will tend to decrease the noncondensable mass in the drywell, thereby minimizing the mixed drywell/wetwell air mass used in the ideal gas law to calculate the minimum containment pressure. The staff finds that for the Pilgrim plant, the initial drywell temperature and humidity are conservative and are therefore acceptable.

3.1.2 Analytical Results

The licensee presented two values of the peak calculated suppression pool temperature. One of these incorporates the ANS 5.1-1979 decay heat with no added uncertainty, and one estimates decay heat with a 2-sigma uncertainty added to the decay heat. For the estimated plus 2-sigma case, the peak calculated suppression pool temperature is 185 °F, whereas the peak calculated suppression pool temperature with no added decay heat uncertainty is 178 °F. The peak calculated suppression pool temperature with a 65 °F service water temperature is 166 °F (current licensing basis). For the case of the current licensing basis, the licensee is not certain which decay heat model was used, but assumes that it was the May-Witt model. One of the new assumptions is the use of the ANS 5.1-1979 decay heat. The staff informed BECo that the

ANS 5.1-1979 decay heat input has not been accepted by the staff and that to assure a conservative decay heat input a 2-sigma uncertainty should be added. As a bounding analysis to support this amendment BECo has estimated the effects of adding 2-sigma uncertainty by changing the suppression pool temperature. BECo has committed to perform the actual 2-sigma uncertainty added analysis within 180 days of issuance of this amendment. The staff considers this a condition for approval of the amendment.

The licensee also presented 2 curves of the calculated containment pressure versus time, again corresponding to the decay heat model used (i.e., estimated plus 2-sigma or nominal ANS 5.1). The calculated pressure increases when the estimated 2-sigma uncertainty is added to the decay heat. However, because of the corresponding increase in suppression pool temperature with the estimated 2-sigma uncertainty added, the margin between the calculated containment pressure and the pressure required for NPSH is approximately the same for the nominal and the plus 2-sigma cases (approximately 6 psi). In either case, the peak pressure calculated in the minimum pressure analysis is approximately 7.5 or 9.9 psig for the decay heat with no uncertainty and with an estimated plus 2-sigma uncertainty, respectively. Both of these pressure profiles bound, with adequate margin, the overpressure requested (1.9 and 2.5 psig) over the time periods for which it is requested.

3.1.3 Benchmark Analysis

The licensee submitted a comparison of the calculated containment pressure and suppression pool temperature produced by the equilibrium methodology used in this license amendment request. The same methodology is used in the current licensing basis, but because a code other than SHEX was used in the current licensing basis, an evaluation to compare, or benchmark, the results produced with the SHEX code and the original code is necessary. When a benchmark analysis is performed using the same initial conditions and assumptions as the original analysis, any differences in the results can be attributed to differences in the code and/or methodology used.

The licensee presented a comparison of recently obtained results with those depicted on Figures 14.5-9 and 14.5-10 of the UFSAR. These figures show the total NPSH available vs. suppression pool temperature, and the containment pressure vs. time, respectively. Figure 14.5-10 also shows suppression pool temperature vs. time. The benchmark case used initial conditions and assumptions that were the same as those used in the original FSAR analysis (i.e., initial pressures, temperatures, humidities, etc.).

With regard Figure 14.5-9, the total NPSH available in the benchmark analysis is essentially identical to that shown on current Figure 14.5-9, with the benchmark case showing about 0.2 ft. greater NPSH available than the original analysis. With regard to Figure 14.5-10, the benchmark analysis indicates a peak minimized pressure of approximately 18 psia, whereas the original analysis results in a peak minimized pressure of approximately 18.8 psia. The suppression pool temperature profile and peak values are virtually identical between the two analyses.

The licensee stated that any differences in the results between the current and the benchmark analysis are probably due to small variations in input. While the staff has not conducted an exhaustive review of what may be the cause of the differences between the benchmark analysis and the original analysis, it notes that small variations in inputs or modeling assumptions made in the different computer codes could reasonably be expected to have an effect on the calculated results. Furthermore, the benchmark analysis tends to predict a lower containment pressure, which is conservative for NPSH. Therefore, the staff finds that the results produced by the benchmark analysis are, within reasonable bounds, comparable to those produced by the original analysis, and that, for the case of Pilgrim, adequate conservatism will be maintained with the current methodology and SHEX code.

The staff has reviewed the licensee's minimum containment pressure and peak suppression pool temperature analyses, conducted for the purpose of calculating the NPSH available to the LPCI and CS pumps and for determining the design-basis peak suppression pool temperature, and finds that the analyses have been conducted using assumptions that are conservative for the particular purpose; i.e., assumptions made to minimize the calculated containment pressure and maximize the calculated suppression pool temperature. Furthermore, the staff finds that the methodology used by the licensee, which incorporates the SHEX computer code, maintains an adequate degree of conservatism relative to the original licensing basis methodology, as shown by the licensee's benchmark analysis. The staff finds that there is reasonable assurance that the requested overpressure will be available for the time periods for which it is requested. Therefore, the licensee's minimum containment pressure analysis and peak suppression pool temperature analysis are acceptable.

3.2 Net Positive Suction Head (NPSH)

3.2.1 Residual Heat Removal And Core Spray NPSH Calculation

The licensee provided evaluations of post-LOCA NPSH for CS and RHR pumps. The RHR and CS pumps are part of the CSCS as described in the PNPS UFSAR Section 6 (Reference 9). Generally, the evaluations are divided into two portions as follows:

Short-Term:	0 to 600 seconds (10 minutes), no operation action credited, vessel injection phase
Long-Term:	600 seconds to completion of event, operator actions credited, containment cooling phase

Section 14.5.3.1.3 in the FSAR established the 600 second mark for operator action and the time at which credit for manual initiation of containment cooling can be taken. Therefore for the long-term case, operator action is credited at the 600 second mark. The staff notes that in both cases, the core spray pumps have a higher NPSH requirement than the RHR pumps. Therefore, NPSH evaluation discussed here are for the CS pumps only.

3.2.2 Short-Term NPSH Requirements

On June 12, 1997 members of the staff met with the licensee regarding the licensing basis amendment request. The staff questioned the licensee as to why the short-term NPSH requirements were not evaluated in the submittal. Recently, the staff has reviewed submittals for containment overpressure credit where the limiting single failure for the short-term was LPCI loop select. The licensee stated that since PNPS is not a ring header plant, CS flow is not affected by a LPCI loop select failure as seen at other facilities. As such, the licensee concluded that adequate NPSH exists during the short-term assuming the calculated amount of debris was instantaneously on the new strainers and no credit for containment overpressure. The staff performed confirmatory calculations using data for 75 °F service water inlet temperature provided in the licensee's calculation M-662 Revision E2 (Reference 10) and the following equation provided in letter dated May 14, 1997.

$$P_c Req'd = P_{vp} + (NPSHR - H_z + H_{sl} + H_{debris}) \frac{\rho}{(144 \frac{inches^2}{feet^2})}$$

where: P_c Req'd = Containment pressure required for adequate NPSH, psia
P_{vp} = Vapor pressure at pool temperature, psia

NPSHR = Net positive suction head required, feet
H_z = Elevation head, feet
H_{sl} = Suction line losses, feet
H_{debris} = Head loss due to debris, feet
ρ = Density of water in pool, lb/ft^3

The staff's analysis concluded that approximately 10.6 psia of containment pressure was required for adequate NPSH for the CS pumps at the 600 second mark. This value is below atmospheric pressure and includes the debris term on the new strainers. This analysis demonstrates that the short-term NPSH requirements are not more limiting than the long-term NPSH requirements.

3.2.3 Long-Term NPSH Requirements

The bounding NPSH case for RHR and CS pumps for long-term evaluation was determined to be a DBA LOCA. The evaluation performed was time and temperature dependent beginning at 102 seconds following a design-basis accident. However, as stated above, credit for operator action is not assumed until the 600 second mark. The peak suppression pool temperature of 177.6 °F was reached at the 19634 second mark. The staff notes that the long-term NPSH requirements were calculated based on a peak suppression pool temperature of 185 °F not 177.6 °F peak pool temperature. The estimated 185 °F temperature is the peak pool temperature derived from ANS 5.1-1979 with a 2-sigma uncertainty added, as requested by the staff. Section 14.5.3.1.2 of the PNPS UFSAR states that the long-term analysis assumes that one RHR loop is available for containment cooling. Therefore, at 600 seconds post-LOCA, the necessary valves are opened to align pumps for the two pump LPCI-Heat Rejection Mode. At 2 hours after the initiation of the accident, a transition from the two pump LPCI-Heat Rejection Mode to a one pump LPCI-Heat Rejection Mode is assumed. The one pump configuration should provide rated heat removal from the containment.

Under this bounding event, the licensee evaluated the long-term NPSH requirements for RHR and CS crediting operator actions and accounting for the new strainers and debris calculated using Regulatory Guide 1.82 Revision 1 (Reference 11). In order to account for increasing service water temperatures in the summer months, the licensee has requested that the current licensing basis be changed from 65 °F to 75 °F. As a result of the increased service water temperature, the licensee requested credit for the following overpressure for specified time periods.

Time Period (seconds)	Containment Overpressure (psig)
0 - 1200	0
1200 - 6000	1.9
6000 - 30 hours	2.5

The staff has reviewed the licensee's minimum pressure analysis which demonstrated the existence of 1.9 and 2.5 psig containment overpressure and finds that adequate overpressure exists for NPSH concerns. Based on this information, the following assumptions were made:

1. The old ECCS strainers were replaced with large capacity stacked disc pump suction strainers for both the CS and RHR pumps during RFO #11.

2. PNPS's current licensing basis does not assume that one of the four torus strainers is 100 percent blocked while the others remained clean. The debris related head loss was calculated using Regulatory Guide 1.82 Rev. 1 and the result of 0.01 feet applied to each new strainer.

3. Operator action will be taken at the 600 second mark to align pumps for the 2 pump LPCI-Heat Rejection Mode and to one pump LPCI-Heat Rejection Mode at the 2-hour mark.

4. The licensee estimated a peak suppression pool temperature of 185 °F based on a service water temperature of 75 °F and decay heat ANS 5.1-1979 with a 2-sigma uncertainty added.

5. Two additional feet of headloss was added to the clean strainer suction line headloss of 2.38 feet for CS and 2.62 feet for RHR to account for instrument reading variations during monthly inservice testing (IST) test measurements of the suction line loss. This assumes that the accident

occurs immediately after the IST test was performed and had indications of some strainer blockage (i.e., 2 feet headloss).

6. A suppression pool pressure of 1.9 psig was assumed to exist from 1200 to 6000 seconds and 2.5 psig from 6000 seconds to 30 hours post-LOCA. As discussed above, the containment analysis has shown that the suppression pool pressure credited will be present following the first 1200 seconds post accident.

Based on the above assumptions, the licensee evaluated the minimum suppression pool pressure (i.e., containment pressure) required for pump protection, assuming NPSH Available (NPSHA) was equal to NPSH required using the equation specified above. The licensee's analysis demonstrated that with the 1 RHR train in 2 pump and 1 pump LPCI-Heat Rejection Mode, a limited amount of NPSH deficit exists for the CS pumps. However, with the minimum suppression pool pressure of 1.9 and 2.5 psig assumed (shown on Figure 2 of Attachment 2 of the June 20, 1997 letter), the NPSH deficit is compensated and long-term cooling is assured.

Based on the above analysis, the staff finds that with credit for containment overpressure of 1.9 psig from 1200 to 6000 seconds and 2.5 psig from 6000 to 30 hours, NPSH for the ECCS pumps will be available to meet the long-term worst-case scenario. This credit for the specified containment overpressure accounts for the increase in service water temperature from 65 °F to 75 °F and the resulting effect on the peak suppression pool temperature. The staff concludes that there is reasonable assurance that plant operation in this manner poses no undue risk to health and safety of the public. In addition, the staff finds it acceptable for the licensee to change the UFSAR to reflect these conditions.

3.2.4 Bulletin 96-03

The staff issued NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors," (Reference 12) identifying that the buildup of debris from thermal insulation, corrosion products, and other particulates on ECCS pump strainers could occur, creating the potential for a common-cause failure of the ECCS, which could prevent the ECCS from providing long-term cooling following a LOCA. The staff has requested that all BWR licensees implement appropriate measures to ensure the capability of the ECCS to perform its safety function following a LOCA. NRC Bulletin 96-03 also requested all licensees to implement these actions by the end of the first refueling outage starting after January 1, 1997.

This time frame for implementation was considered appropriate by the staff based on recent cleaning of suppression pools, operator training and appropriate emergency operating procedures, alternate water sources, and a low probability of the initiating event. The licensee has stated that the suppression pool was cleaned of accumulated debris consisting of sludge and corrosion particles during RFO #11. In the case of Pilgrim, containment overpressure is not required, during the first 1200 seconds, for the ECCS pumps to meet the requirements of 10 CFR 50.46(a)(1)(i) with the original licensing basis. The staff notes that this conclusion is based on the licensee's analysis of debris calculated by Regulatory Guide 1.82, Revision 1 and does not take into account the potential for additional blockage as identified in NRC Bulletin 96-03. Appropriate corrective actions, if any, resulting from the licensee's evaluation of NRC Bulletin 96-03 will be implemented in accordance with 10 CFR Part 50 Appendix B. This action will resolve the staff's outstanding questions relative to ECCS performance and will provide long-term assurance that the requirements of 10 CFR 50.46 are met. The resolution of NRC Bulletin 96-03 will be addressed under a separate cover.

3.3 Emergency Diesel Loading Profile Change

The current UFSAR analysis for design basis events and the performance of emergency cooling equipment uses a constant ultimate heat sink temperature of 65 °F at the SSW inlet. Recently, the licensee has performed a design verification of the plant safety analysis using a 75 °F SSW inlet temperature. Since the design basis LOCA analysis using the 75 °F SSW temperature assumes a different core and containment cooling method, it will change the emergency diesel generator (EDG) loadings as follows:

- With a 65 °F SSW inlet temperature, one RHR pump is shut off and one additional SSW system pump and RBCCW system pump are started approximately 10 minutes after the accident in order to support containment cooling initiation.

- With a 75 °F SSW inlet temperature, the above RHR pump that would have been shut off after 10 minutes will now remain operating for 2 hours and the above SSW and RBCCW pumps that start 10 minutes after the accident will remain operating for 2 hours. Thus, the 75 °F case will extend the electrical load profile from 10 minutes to 2 hours. The electrical loads for RHR, SSW, and RBCCW are 639 kW, 83 kW, and 54 kW, respectively.

However, the licensee's submittal did not include the EDG loading calculation that assumes the design-basis LOCA analysis using a SSW inlet temperature of 75 °F. The staff requested and obtained the revised EDG Loading Calculation No. PS-79, dated January 16, 1996.

The staff has reviewed the calculation and finds that among two EDGs (A and B) at Pilgrim, the most heavily loaded diesel is "B," with a steady state load of 2563 kW (including the above 639-kW RHR pump) in the period of time between accident initiation and 10 minutes. The calculation assumes that the maximum EDG load would occur when additional SSW (83 kW) and RBCCW (54 kW) pumps are starting at 10 minutes after essential valves have cycled. This would load EDG B to 2700 kW, but the loading is still below the 2000-hour/yr limit of 2750 kW and the 2-hour/yr limit of 3000 kW.

Based on our review of the licensee's EDG loading calculation analyzed for the design-basis LOCA using 75 °F SSW temperature, the staff concludes that the EDGs at Pilgrim would not be overloaded and remain within their rating limits. In the calculation, the licensee also stated that the Pilgrim Plant Procedure 2.2.8 has been revised to provide instructions for the operator to monitor EDG loadings to remain within its rated limit.

3.4 Components cooled by RBCCW

Each RHR pump is equipped with a seal cooler (supplied by RBCCW) which lowers the temperature of the water injected into the seal chamber to flush and cool the rotating seal faces so that there will be no flashing at the seal faces. By maintaining the temperature of the seal chamber below saturation, flashing at the seal face is prevented. Flashing is only a concern when the RHR pumps are used in the shutdown cooling (SDC) mode when the RHR system water can exceed 300 °F. The seal cooler is not essential for LPCI or containment cooling since the temperature of the fluid pumped is well below the saturation temperature. Consistent with this basis, the CS pumps have a similar mechanical seal and do not have a seal cooler since the fluid handled by these pumps is either relatively cool water from the condensate storage tank (CST), or suppression pool water at the same temperature as that handled by the RHR pumps during transient and accident scenarios. Therefore, the RHR pumps can still perform their safety function following design basis accidents with a SSW system temperature of 75 °F. In its May 14, 1997 submittal, the licensee also verified that flashing will not occur at the RHR pump seals during the SDC mode as a result of the higher UHS temperatures. Therefore, the increase in UHS temperature is acceptable for cooling the RHR pumps.

The CS pumps do not include a seal cooler but each pump motor has a cooling coil (supplied by RBCCW) immersed in the oil bath provided to lubricate the motor thrust bearings. According to the May 14, 1997 submittal, the maximum peak RBCCW temperature (98 °F) [based on the increased UHS temperature] does not result in exceeding the oil temperature design limits of the CS pump motors. Thus, the operability of the CS pump motors are not affected by the proposed UHS temperature increase.

In its May 14, 1997 submittal, the licensee also identified that all safety-related equipment within the MCC (motor control center) enclosures and the building compartments served by area coolers cooled by RBCCW have been evaluated for the effects of the maximum UHS temperature of 75 °F, and the temperatures of all equipment within the applicable locations have been determined to remain within the equipment design limits. Therefore, the proposed UHS temperature increase is acceptable from the standpoint of the effects on ventilation area coolers affected by the proposed change.

Based on its review, the staff has concluded that the proposed increase in UHS temperature to 75 °F does not result in exceeding the design limits of any safety-related equipment directly or indirectly cooled by the RBCCW system. The proposed changes are, therefore, acceptable with respect to equipment and area cooling. As service water temperature is an important plant parameter that is critical to safe operation and accident mitigation the licensee has proposed to submit an ultimate heat sink TS by the first quarter of 1998 and change the administrative limit from 68 °F to 75 °F. The staff considers this acceptable and a condition of the amendment.

3.5 Equipment Qualification

By letter dated January 20, 1997, the licensee indicated that some of the environmental conditions (e.g., radiation, pressure, temperature, humidity, etc.) to which equipment must be qualified are affected by the 65 °F to 75 °F increase in the SSW injection temperature and were further affected by an error identified in a previous analysis. The licensee: (1) re-evaluated the affected conditions to establish a new environmental qualification accident profile; and (2) reviewed and updated all 10 CFR 50.49 document files to demonstrate qualification to the new accident profile. The licensee's review identified five motor-operated valves located inside the drywell and the containment electrical penetration assemblies for which additional information (test results) was required to complete document files.

In response to an NRC RAI, the licensee by letter dated May 14, 1997, indicated: (1) that during refueling outage #11, the drywell spray flowrate was increased to a minimum design value of 1250 gpm from its previous design value of 720 gpm; (2) the 1250 gpm flowrate completely offsets the drywell temperature increase that would have otherwise resulted from a proposed revision to the containment analysis; (3) all equipment environmental qualification test profiles, thus, continue to envelop the new accident and post-accident profiles; (4) for post-LOCA qualification, Arrhenius methodology continues to be utilized; and (5) containment electric penetrations continue to be qualified to the higher SSW temperature but do not have to meet the additional qualification requirements of 10 CFR 50.49.

Based on the licensee's assessment that environmental qualification test profiles envelop the new accident profiles for 75 °F SSW injection temperature and are qualified, the staff concluded that there is reasonable assurance that electrical equipment will function as required during accident conditions and that electrical equipment meet the requirements of 10 CFR 50.49; however, a more focussed review was considered to be needed for: (1) equipment which utilizes the Arrhenius methodology for post-LOCA qualification; and (2) electrical containment penetrations. The staff's evaluation of these two areas is described in the following sections.

3.5.1 Equipment Which Utilizes the Arrhenius Methodology For Post-Loca Qualification

BECo utilizes the Arrhenius methodology to demonstrate that post-accident operating time is acceptable using test data of a shorter duration than the PNPS specific accident profiles, but having higher temperatures than the required accident conditions. The use of Arrhenius methodology to support qualification of equipment for LOCA and/or longer term post-LOCA environments has not been specifically endorsed by NRC Regulatory Guide, has not been generally accepted, by itself, to demonstrate qualification of equipment in post-LOCA environments, and has not been validated by test. Therefore, the staff has maintained that the use of Arrhenius methodology, by itself, without supporting justification or technical basis, is not considered an acceptable approach for supporting qualification of electric equipment for LOCA environments.

Electric Power Research Institute's (EPRI's), Nuclear Power Plant Equipment Qualification Reference Manual indicates that the Arrhenius method has been employed to relate accident test temperatures to postulated accident temperatures. If the Arrhenius model and activation energy value are applicable to the test and accident temperatures, then the model may arguably be used in various ways to draw correlations between the accumulated thermal damage occurring during various phases of LOCA testing. This approach has been used principally to support long-term operability in post-LOCA environments when it is desirable to have a test duration that is shorter than the actual required operability time. For example, the test temperature plateau dropped to 212 °F at 5 days into the 30-day test. The required post-LOCA temperature dropped to 190 °F after 5 days and remained constant for an additional 175 days. Thus, although the test temperature envelopes the required post-LOCA temperature, it lasts only 25 days and not 175 days. It is a common practice to argue that the higher test temperature (212 °F) can be viewed as an accelerated version of the actual post-LOCA temperature (190 °F). After using Arrhenius methods to determine equivalent degradation for 25 days at 212 °F and 175 days at 190 °F, if it turns out the equivalent degradation for 25 days at 212 °F is greater than 175 days at 190 °F, it can be argued that the test is conservative with respect to the actual post-LOCA conditions.

The staff agrees that the Arrhenius model and activation energy value can be shown to be applicable to the test and accident temperatures. The Arrhenius methodology has been typically used and accepted to project degradation (aging) under constant temperature conditions. In the example, the initial stage (e.g., the first 5 days) of the LOCA temperature profile -- which are not constant -- have been excluded from the Arrhenius calculation. The test conditions from 5 to 30 days and the accident profile from 5 to 180 days (175 days) for which the Arrhenius calculation is being applied are relatively constant. If temperature conditions are constant and if LOCA conditions do not cause material change, the staff agrees that the Arrhenius model and activation energy value can be shown to be applicable to support long-term operability in post-LOCA environments based on short-term testing at higher temperature.

The licensee indicated by telephone discussions and subsequently by letter dated June 20, 1997, that they and their material consultant have concluded that using the Arrhenius methodology is a valid approach due to the temperature rating of the materials used. Based upon published technical information, fundamental considerations of polymer science and chemistry, direct experience with the materials, and the material consultant's experience with polymeric materials and the aging characteristics of these materials, it was concluded (for all material located in the drywell at PNPS) that the material properties would not change such that the use of the Arrhenius methodology would be invalid in a post-LOCA environment. For nuclear plant containment applications virtually all elastomers and most thermoplastics are cross-linked. The cross-linking process not only eliminates polymer melting, but increases its heat aging resistance, chemical resistance, and physical durability under a wide range of conditions. Accordingly, the materials can be used at high temperatures without undergoing chemical change. The licensee, therefore, concluded that the Arrhenius methodology is acceptable to use after the equipment has been subjected to LOCA temperatures.

For PNPS, Arrhenius methodology has been utilized for those cases where the test profile does not envelop the accident profile for the required duration (i.e., 30 days + 10 percent margin or 33 days). To determine degradation, the licensee indicated that the Arrhenius equation has been applied starting at time zero and includes the LOCA transient as part of the calculation for degradation. If an additional peak transient is utilized to assure performance margin during testing, the licensee indicated that this additional transient is not applied to calculate degradation.

As implied by the above described industry guidelines, and based on discussion with others familiar with the application of the Arrhenius methodology, the staff has concluded that the Arrhenius model and activation energy value are generally not considered to be an accurate methodology for establishing degradation of equipment during transient temperature conditions i.e., the initial stage of the LOCA which are not constant. Thus, the staff disagrees with PNPS application of the Arrhenius methodology during transient temperature conditions and its utilization as part of their process for assuring qualification of electrical equipment in post-LOCA environments.

In response to the above described disagreement, the licensee indicated by June 20, 1997, letter that they have reviewed all in-containment EQ equipment utilizing the Arrhenius methodology starting with its application at times greater than 1 hour past the transient peak temperature. The review focused on determining whether all in-containment equipment would still be qualified with a test margin two times greater than the 10 percent margin required by IEEE 323-1974. The licensee concluded that all equipment, except Rockbestos wire associated with Limitorque limit switches, meets the two times the 10 percent margin criteria. The Rockbestos wire was determined to have a 17 percent margin.

Based on the licensee's assessment (described above) that Rockbestos wire is qualified with a 17 percent margin and all other equipment is qualified with a margin two times greater than the 10 percent margin required by IEEE 323-1974, it appears that equipment has sufficient additional margin above that which is required to compensate for any uncertainties associated with the application of the Arrhenius methodology in post-LOCA environments. In addition, the licensee's application of the Arrhenius methodology during transient temperature conditions does not appear to have a significant impact on qualification of equipment. Thus, the licensee's assessment provides reasonable assurance that equipment required to meet the requirements of 10 CFR 50.49 will function as required during accident conditions with the higher SSW inlet temperature of 75 °F and is considered qualified. However, as a separate initiative outside the scope of this evaluation, the NRC staff will continue to review this type of analytical methodology in order to assure that the approach used was appropriate and conservative.

3.5.2 Electrical Containment Penetrations

In response to an NRC RAI, the licensee by letter dated May 14, 1997, indicated that containment penetrations (either electrical or piping) are considered extensions of containment. They are considered mechanical devices and therefore not subject to the requirements of 10 CFR 50.49. Containment penetrations that contain cables that power equipment required to mitigate the consequences of an accident are required to be qualified to the criteria specified in 10 CFR 50.49 only to the extent that failure of the penetration will not affect operability of safety-related cables. Qualification of the penetration to assure its safety function (i.e., containment integrity) is not subject to the requirements of 10 CFR 50.49. Penetrations that contain non-safety-related cable are similarly not subject to the requirements of 10 CFR 50.49.

The staff disagrees. Containment penetrations which contain either safety or non-safety-related circuits perform a safety function to maintain containment integrity. Containment penetrations should be considered safety-related electrical equipment. If failure of the containment electric penetration can cause loss of safety function (i.e., containment integrity), paragraph (b)(1)(iii or C) of 10 CFR 50.49 requires that the containment electric penetration be covered by 10 CFR 50.49.

The licensee has submitted an evaluation that concludes that failure of cable that passes through primary containment would not adversely affect the primary containment boundary because: (1) current limiting devices have been installed in the electrical circuit so that any potentially damaging fault currents will be interrupted prior to loss of the penetration's safety function (i.e., containment integrity); and (2) the electrical penetrations have been environmentally qualified (i.e., tested at accident environments while being subject to fault current) to demonstrate their capability for maintaining safety function--containment integrity. Therefore, the staff has concluded that the licensee's evaluation provides reasonable assurance that containment penetrations which contain electrical circuits will function as required during accident conditions at the higher SSW temperature.

The staff, however, believes that containment electrical penetrations should be considered electric equipment and should therefore be covered by 10 CFR 50.49. The staff also believes that the existence of current limiting devices and qualification do not provide an appropriate argument for excluding electrical penetrations from being covered by 10 CFR 50.49.

4.0 SUMMARY

The staff has concluded that sufficient containment pressure exists post LOCA to assure that the NPSH available is greater than the NPSH required. This analysis included several changes in initial conditions assumed for the containment analysis, the most significant ones being the change to the decay heat input and service water temperature. The staff has also reviewed the effects of these changes on EDG loadings, EQ, and any safety-related equipment directly or indirectly cooled by the RBCCW system and found the equipment to be operable. The staff, therefore, authorizes BECo to change the UHS administrative limit from 68 °F to 75 °F, and change the Updated Final Safety Analysis Report (UFSAR) to reflect the use of containment pressure to compensate for the deficiency in NPSH following a design basis accident and increase the accident analysis design UHS temperature from 65° F to 75° F. As part of this amendment, Boston Edison Company (BECo/licensee) has proposed to submit a Technical Specification amendment for the UHS temperature by the first quarter of 1998. In addition, within 180 days of issuance of this amendment, BECo has committed to complete the containment analysis using the ANS 5.1-1979 Decay Heat Curve with a 2-sigma uncertainty added. The staff considers BECo's commitments acceptable and has conditioned the amendment accordingly.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Massachusetts State official was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (62 FR 8792). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

1. Boulette, E. T., Boston Edison Company, to USNRC, "Request for Review," January 20, 1997.

2. Olivier, L. J., Boston Edison Company, to USNRC, "Significant Hazards Evaluation for Pilgrim Nuclear Power Station's Net Positive Suction Head Analyses," January 30, 1997.

3. Boulette, E. T., Boston Edison Company, to USNRC, "Supplemental Submittal on Pilgrim Station NPSH Analysis," February 27, 1997.

4. Boulette, E. T., Boston Edison Company, to USNRC, "Revised Request for License Amendment to Credit Containment Pressure in ECCS NPSH LOCA Analyses," April 11, 1997.

5. Olivier, L. J., Boston Edison Company, to USNRC, "Response to Request for Additional Information Concerning Pilgrim Station Crediting Containment Overpressure in the Net Positive Suction Head Analysis for the Emergency Core Cooling Pumps," May 14, 1997.

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8. Milano, P. D., USNRC, to R. W. Cooper, USNRC, "Request for Technical Assistance Regarding the Pilgrim Licensing Basis and Subsequent Modification Under 50.59 for Two Safety Significant Problems," February 12, 1997.

9. Pilgrim Nuclear Power Station Final Safety Analysis Report, Revision 20, February 19, 1997.

10. Pilgrim Nuclear Power Station Calculation M-662 Revision E2, "RHR and Core Spray Pump NPSH and Suction Pressure Drop," April 10, 1997.

11. Regulatory Guide 1.82, Revision 1, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," USNRC, November 1985.

12. NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Stainers By Debris in Boiling-Water Reactors," May 6, 1996.

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