Docket Nos.50-293

February 5, 1982

Mr. A. Victor Morisi, Manager
Nuclear Operations Support Department
Boston Edison Company
M/C Nuclear
800 Boylston Street
Boston, Massachusetts 02199

Dear Mr. Morisi:

The Commission has issued the enclosed Amendment No. 53 to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station. This amendment consists of changes to the Technical Specifications in response to your application dated January 15, 1982.

These changes to the Technical Specifications reflect modifications made as part of the Mark I Containment Long Term Poogram. These modifications involve shortening of the ventheader downcomers and necessitate a decrease in the drywell-suppression chamber differential pressure.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

ORIGINAL SIGNED BY

Kenneth T. Eccleston, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 53 to DPR-35
2. Safety Evaluation
3. Notice of Issuance

cc: w/encl.
See next page

Distribution:	Docket File	NRC PDR	Local PDR	ORB#2 Rdg	D. Eisenhut
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no legal objection subject to comments on FR notice and SER p 1

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OFFICIAL RECORD COPY

Mr. A. Victor Morisi
Boston Edison Company

cc:

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

BOSTON EDISON COMPANY

DOCKET NO. 50-293

PILGRIM NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 53
License No. DPR-35

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

   A. The application for amendment by Boston Edison Company (the licensee) dated January 15, 1982, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   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 amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B of Facility Operating License No. DPR-35 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 53, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

D B Vassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 5, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 53

FACILITY OPERATING LICENSE NO. DPR-35

DOCKET NO. 50-293

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment Number and contain a vertical line indicating the area of change.

Remove	Replace
152A	152A
166	166
171	171

3.7 CONTAINMENT SYSTEMS (Cont'd)

h. During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig at normal cool down rates if the pool temperature reaches 120°F.

i. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.17 psid, except as specified in j and k.

j. The differential pressure shall be established within 24 hours of placing the reactor in the run mode following a shutdown. The differential pressure may be reduced to less than 1.17 psid 24 hours prior to a scheduled shutdown.

k. The differential pressure may be reduced to less than 1.17 psid for a maximum of four (4) hours for maintenance activities on the differential pressure control system and during required operability testing of the HPCI system, the relief valves, the RCIC system and the drywell-suppression chamber vacuum breakers.

l. If the specifications of Item i, above, cannot be met, and the differential pressure cannot be restored within the subsequent six (6) hour period, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition in twenty-four (24) hours.

m. Suppression chamber water level shall be maintained between -6 to -3 inches on torus level instrument which corresponds to a downcomer submergence of 3.00 and 3.25 feet respectively.

4.7 CONTAINMENT SYSTEMS (Cont'd)

e. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift when the differential pressure is required.

f. Suppression chamber water level shall be recorded at least once each shift when the differential pressure is required.

BASES:

3.7.A & 4.7.A Primary Containment

The integrity of the primary containment and operation of the core standby cooling system in combination limit the off-site doses to values less than those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception is made to this requirement during initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required. There will be no pressure on the system at this time, thus greatly reducing the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth such that a rod drop would not result in any fuel damage. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offer a sufficient barrier to keep off-site doses well below 10 CFR 100 limits.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1035 psig. Since all of the gases in the drywell are purged into the pressure supression chamber air space during a loss-of-coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 45 psig which is below the maximum of 62 psig. Maximum water volume of 94,000 ft^3 results in a downcomer submergency of 4'-0" and the minimum volume of 84,000 ft^3 results in a submergence approximately 12-inches less. Mark I Containment Long Term Program Quarter Scale Test Facility (QSTF) testing at a downcomer submergency of 3.25 feet and 1.17 psi wetwell to drywell pressure differential shows a significant suppression chamber load reduction and Long Term Program analysis and modifications are based on the above submergence and ΔP.

Should it be necessary to drain the suppression chamber, provision will be made to maintain those requirements as described in Section 3.5.F BASES of this Technical Specification.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the pressure suppression pool is maintained below 160°F during any period of relief-valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high pressure suppression chamber loadings.

BASES:

3.7.A & 4.7.A Primary Containment (Cont'd)

The primary containment is normally slightly pressurized during periods of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase oxygen concentration. Once the containment is filled with nitrogen to the required concentration, no monitoring of oxygen concentration is necessary. However, at least twice a week the oxygen concentration will be determined as added assurance. Mark I Containment Long Term Program testing showed that maintaining a drywell to wetwell pressure differential to keep the suppression chamber downcomer legs clear of water significantly reduced suppression chamber post LOAC hydrodynamic loads. A pressure of 1.17 psid is required to sufficiently clear the water legs of the downcomers without bubbling nitrogen into the suppression chamber at the 3.00 ft. downcomer submergence which corresponds to approx. 84,000 ft.^{3} of water. Maximum downcomer submergence is 3.25 ft. at operating suppression chamber water level. The above pressure differential and submergence number will be used in the Pilgrim I Plant Unique Analysis to be submitted to the NRC.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 53 TO FACILITY LICENSE NO. DPR-35

BOSTON EDISON COMPANY

PILGRIM NUCLEAR POWER STATION

DOCKET NO. 50-293

Principal Author: Kenneth T. Eccleston

1.0 Introduction

By letter dated January 15, 1982, Boston Edison Company (the licensee) requested an amendment of Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station (the facility). The proposed amendment would reduce the maximum downcomer submergence to 3.25 feet and would reduce the minimum drywell-torus differential pressure from 1.5 psid to 1.17 psid. The licensee is shortening the length of the downcomers as part of the Mark I Containment Long Term Program (LTP) and has determined that a change in the Technical Specification requirements for drywell-torus differential pressure is necessary.

2.0 Evaluation

The purpose of the Mark I Containment Long Term Program is to perform a complete reassessment of the suppression chamber (torus) design to include suppression pool hydrodynamic loads, which were neglected in the original design, and to restore the originally intended design safety margins of the structure.

NUREG-0661 describes the generic techniques for the definition of suppression pool hydrodynamic loads in a Mark I system and the related structural acceptance criteria.

One method of suppression pool hydrodynamic load mitigation that the Mark I Owners Group has adopted for the LTP is reducing the initial submergence of the downcomer in the suppression pool to a minimum of at least three feet. By shortening the length of the downcomer, the pool volume (i.e., thermal capacity) of the original design would be maintained. This approach, however, raises concern regarding the increased potential for uncovering the downcomers and steam condensation capability, both of which could lead to torus overpressurization.

2.1 Seismic Slosh

The potential for downcomer uncovery is addressed in the assessment of seismic slosh. This assessment was performed at the most extreme conditions that could potentially lead to uncovering of the downcomers and was predicted on a minimum three-foot downcomer submergence.

Seismic motion induces suppression pool waves which can (1) impart an oscillatory pressure loading on the torus shell, and (2) potentially lead to uncovering the ends of the downcomers, which could result in steam bypass of the suppression pool and potential overpressurization of the torus, should the seismic event occur in conjuction with a Loss of Coolant Accident (LOCA). To assess these effects, the Mark I Owners Group undertook the development of an analytical model which would provide plant-specific seismic wave amplitudes and torus wall pressures. This model was based on 1/30-scale "shake test" data for a Mark I torus geometry.

Based on the results of plant-specific analyses, using the analytical model, the Mark I Owners Group concluded that (1) the seismic wave pressure loads on any Mark I torus are insignificant in comparison with the other suppression pool dynamic loads, and (2) the seismic wave amplitudes will not lead to uncovering the downcomers for any Mark I plant. This conclusion was based on the maximum calculated pressure loads and the minimum wave trough depth relative to the downcomer exit.

We have reviewed comparisons of the analytical predictions with scaled-up test data, the small-scale test program, and the seismic spectrum envelope used in the plant-specific analyses. Based on this review, we conclude that the seismic slosh analytical predictions will provide reasonably conservative estimates of both the wall pressure loading and the wave amplitude, for the range of Mark I plant conditions.

Since the maximum local wall pressures were found to be less than 0.8 psi at a 95% upper confidence limit, the Mark I Owners Group has proposed that the seismic slosh loads may be neglected in the structural analysis. We agree that the seismic slosh loads are insignificant in comparison with the other suppression pool dynamic loads. On this basis, we conclude that neglecting seismic slosh loads for the plant-unique analyses is acceptable.

The results of the slosh wave amplitude predictions indicate that, within the local area of maximum amplitude and with maximum suppression pool drawdown (resulting from ECCS system flows), the slosh waves will not cause uncovering of the downcomers. We have reviewed the assumptions used in these analyses and conclude that they are sufficiently conservative. Based on the above discussion, we find the proposed change acceptable.

2.2 Condensation Capability

Condensation capability of the suppression pool is a function of the local pool temperature in the vicinity of the downcomer exit. Full Scale Test Facility (FSTF) test results and foreign test data have shown that thermal stratification occurs, and becomes more severe as the downcomer submergence is reduced. The most severe thermal stratification has been observed in low flow tests with a quiescent pool. However, in actual plant conditions, the Residual Heat Removal (RHR) system and Safety Relief Valve (SRV) discharge provide sufficient long-term pool mixing to minimize thermal stratification. Even with vertical thermal stratification, we have determined that the high energy reposition is accompanied by an increased flow and mixing, which prevent overpressurization of the torus. In addition, the analytical predictions of the torus pressure and bulk temperature response have been found to be conservative when compared with FSTF test data for plant simulated initial conditions. The local temperature variation in the pool which has been observed in the test data is not significant to the structure, and therefore, need not be considered in the structural analysis.

Based on this assessment, we conclude that a minimum initial downcomer submergence of three feet is acceptable, and there is sufficient conservatism in the containment response analysis techniques to accommodate the effects of thermal stratification.

2.3 Differential Pressure

The introduction of a positive pressure differential between the drywell and the suppression chamber air volume reduces the height of the water leg inside the downcomer. The reduced water leg permits the downcomers to clear earlier in the LOCA transient with the drywell consequently at a lower pressure. This effect reduces both the downward and upward pressure loads on the containment. Mark I Containment LTP Quarter Scale Test Facility testing at a downcomer submergence of 3.25 feet and a 1.17, psi drywell to wetwell pressure differential shows a significant suppression chamber load reduction.

The length of the water leg inside the downcomer is limited by the downcomer submergence. Due to the shortening of the downcomers the drywell to torus differential pressure will be reduced by an amount equivalent to the reduction (9") in the water leg inside the downcomer. Long Term Program analysis and modifications are based on a downcomer submergence of 3.25 feet and a drywell to torus differential pressure of 1.17 psi. The licensee will submit for post-implementation review the Plant Unique Analysis for Pilgrim utilizing this pressure differential and downcomer submergence value to the NRC.

Since this modification will reduce the suppression pool hydrodynamic loads and act to restore the orignally intended margins of safety, the proposed modification is acceptable. Therefore, we find the proposed Technical Specification changes acceptable.

3.0 Environmental Considerations

We have determined that the amendment does not involve a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR Section 51.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendment.

4.0 Conclusions

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: February 5, 1982

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-293

BOSTON EDISON COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 53 to Facility Operating License No. DPR-35 issued to the Boston Edison Company, which revises the Technical Specifications for operation of the Pilgrim Nuclear Power Station located in Plymouth County, Massachusetts. The amendment is effective as of its date of issuance.

The amendment modifies the Technical Specifications to reflect modifications associated with the Mark I Containment Long Term Program, specifically regarding shortening of the ventheader downcomers.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR 51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated January 15, 1982, (2) Amendment No. 53 to License No. DPR-35, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C. and at the Plymouth Public Library, North Street, Plymouth, Massachusetts 02360. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland this 5th day of February 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing