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MAR 20 1982

Docket No. 50-293

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. 58 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 March 1, 1982.

These changes to the Technical Specifications incorporate limiting conditions for operation and surveillance requirements for drywell temperature and drywell temperature monitoring instrumentation.

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. 58 to DPR-35
2. Safety Evaluation
3. Notice

cc w/enclosures
See next page

No legal objection to form of amend. & FR Notice

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OFFICE	ORB#2	ORB#2	ORB#2	AD:OR:DL	OELD		
SURNAME	SNorris	KEccleston:pob:MC	DVassallo	TNovak			
DATE	3/19/82	3/19/82	3/19/82	3/ /82	3/24/82		

NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—335-960

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. 58
License No. DPR-35

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

   A. The application for amendment by the Boston Edison Company (the licensee) dated March 1, 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. 58, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 20, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 58

FACILITY OPERATING LICENSE NO. DPR-35

DOCKET NO. 50-293

Revise Appendix A as follows:

Add the following new pages:

44b
44c
59b
66b
73a

3.2.H Drywell Temperature

1. The drywell temperature shall be maintained within the following limits when the reactor coolant temperature is above 212°F.

   - Above elevation 40' : ≤194°F

   Equal to or Below elevation 40' : ≤150°F

   Upon determination that the drywell temperature at any elevation has exceeded the above limits, the drywell temperature at each elevation shall be logged every 30 minutes. The drywell temperature shall be reduced to within the limits within 24 hours; otherwise, corrective action shall be as specified in 3.2.H.2, 3.2.H.3.

2. If the drywell temperature has exceeded either limit of 3.2.H.1 for greater than 24 hours, an engineering evaluation shall immediately be initiated to assess potential damage and render a determination of ability of safety related equipment to perform its intended function.

   If either limit of section 3.2.H.1 has been exceeded for greater than 24 hours, further action to justify continued operation shall be determined by an engineering evaluation. which must be completed within one week.

3. If the requirements of 3.2.H.2 have not been met an orderly Shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

4. If the drywell temperature at any elevation exceeds 215°F and the temperature cannot be reduced to below 215°F within 30 minutes a reactor shutdown shall be initiated and the reactor shall be in cold shutdown condition within 24 hours.

4.2.H Drywell Temperature

1. When reactor coolant temperature is above 212°F, the drywell air temperature limits will be determined by reading the instruments listed in Table 3.2.H. These instruments shall be logged once per shift, and each reading compared to the limits of Section 3.2.H.1.

2. Instrumentation shall be calibrated and checked as indicated in Table 4.2.H.

5. The limiting conditions of operation for the instrumentation that monitors drywell temperature are given in Table 3.2.H.

PNPS
TABLE 3.2.H

Drywell Temperature Surveillance Instrumentation

Minimum # of Operable Elements	Instrument #	Nominal Instrument Elevation	Type	Note
Above 40 Feet Elevation				
1/ELEV	TE-5050A 1/2 TE-5050B 1/2	80' 80'	RTD	(1) (2) (4)
1/ELEV	TE-5050C 1/2 TE-5050D 1/2	87' 87'	RTD	(1) (2) [illegible]
1/ELEV	TE-5050E 1/2 TE-5050F 1/2	60' 60'	RTD	(1) (2) (4)
Below 40 Feet Elevation				
1/ELEV	TE-5050G 1/2 TE-5050H 1/2	41'	RTD	(1) (3) (4)
1/ELEV	TE-5050J 1/2 OR TE-5050K 1/2	32'	RTD	(1) (3) (4)

Notes:

1. The 5050 series temperature elements are dual-elements
2. At least one element of one RTD on each elevation shall be operable.
3. At least one element of one RTD on elevation 41 and one element of one of the RTD's at nominal elevation 32 shall be operable.
4. If the minimum number of operable RTD's as specified in Note 2 and 3 above are not availiable and cannot be made available within 24 hours, an orderly shutdown shall initiated and the reactor shall be in a Cold Shutdown condition within 24 hours of shutdown initiation.

PNPS
TABLE 4.2.H

Minimum Test & Calibration Frequency for Drywell Temperature Surveillance Instrumentation

Instrument Channels/ Nominal Elevation	Calibration Frequency	Instrument Check
80 Feet	Each Refueling Outage	Once per Shift
87 Feet	Each Refueling Outage	Once per Shift
60 Feet	Each Refueling Outage	Once per Shift
41 Feet	Each Refueling Outage	Once per Shift
32 Feet	Each Refueling Outage	Once per Shift

3.2

Drywell Temperature

The dyrwell temperature limitations of Specification 3.2.H.1 ensure that safety related equipment will not be subjected to excess temperature. Exposure to excessive temperatures may degrade equipment and can cause loss of its operability.

The temperature elements for monitoring drywell temperature specified in Table 3.2.H were chosen on the basis of their reliability, location, and their redundancy (duel - element RTD's). These temperature elements are the primary elements used for the PCILRT.

The temperature limits specified in 3.2.H.1 are based on the BECo report entitled Drywell Temperature Report, dated January 28, 1982. The limits derived from this report take into consideration the long-term effects of ambient temperature on equipment design limits and material degradation of components required for accident mitigation or plant shutdown. The evaluation process addressed the actual assessment of potential damage and the determination of equipment status from the standpoint of both qualification integrity (for safety-related equipment) and reliability to perform it's intended function.

If the drywell temperature exceeds the limits specified in 3.2.H.1 an engineering evaluation must be initiated in order to determine whether any safety related component has been adversely affected.

The limiting drywell temperature value of 215^{o}F (Section 3.2.H.2) was selected as to guarantee that ECCS trips occur on/or before present Technical Specification values.

The time interval of 30 minutes between successive drywell temperature instrument readings (Section 3.2.H.1) was selected so as to guarantee that ECCS trips occur on/before present Technical Specification values in the event of a drywell temperature excursion in excess of 215 F.

The instrument check interval of once per shift provides adequate assurance of equipment operability based upon engineering judgement.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 58 TO FACILITY LICENSE NO. DPR-35

BOSTON EDISON COMPANY

PILGRIM NUCLEAR POWER STATION

DOCKET NO. 50-293

Authors: K. Eccleston
H. Garg
O. Rothberg

I. Introduction

During previous cycles of operation, Pilgrim Nuclear Power Station was operated with higher than normal ambient drywell temperatures due to inadequate drywell cooling. The temperatures experienced during Cycle 5 operation ranged as high as 250°F at the upper elevations of the drywell. This evaluation addresses the effects of these higher than normal drywell temperatures on structures and equipment important to safety and the adequacy of the Technical Specification changes proposed by Boston Edison Company (the licensee) in its March 1, 1982 application to provide limiting conditions for operation and surveillance requirements concerning drywell temperature and associated temperature monitoring instrumentation.

II. Background

On September 26, 1981 during a routine shutdown for refueling, the installed Yarway water level instrumentation experienced oscillations. These oscillations have been attributed to flashing in the reference leg of these Yarway instruments caused by the excessively high drywell temperatures. By letter dated January 18, 1982 the licensee provided its evaluation of the effects of high drywell temperature operation on structures, components, and on transient and accident analyses. This evaluation also described the measures taken to correct or repair identified deficiencies and described the licensee's plans and programs for modifications and replacements to enhance drywell cooling capability and to assure equipment operability and qualification for at least one more cycle of power operation.

III. Evaluation

A. Structural

1. Drywell penetrations

Drywell penetrations are designed for thermal expansion at a drywell temperature of 281°F, which envelopes the drywell temperatures experienced prior to and during the last cycle. Consequently, no detrimental effects of thermal

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expansion on drywell penetrations would have resulted from operation with the maximum drywell temperatures experienced during previous cycles of operation.

2. Differential expansion of the steel and concrete portions of the drywell

The steel containment liner is mechanically attached to the concrete shell only in the lower regions of the drywell (lower 7 feet of the drywell). At these elevations the experienced drywell temperature was below the maximum drywell design temperature; therefore, no adverse effects resulting from differential expansion of concrete and steel would have resulted from operation during previous cycles.

3. Resistance of the drywell structure to jet impingement and other LOCA loads

The Pilgrim Final Safety Analysis Report (Appendix L) containment stress analysis performed assumed target area temperatures of 300°F, which is the same temperature as that of the impinging jet. Since the drywell temperature did not exceed 300°F at any time, these analyses are not invalidated by the higher drywell temperatures experienced during previous cycles.

4. Effect of high drywell temperatures on drywell concrete strength

The licensee has provided the results of analyses performed by its consultant using published test data to determine residual concrete strength after cyclic heating. The licensee states that these analyses indicate an average residual strength of the concrete closest to the liner of approximately 4500 psi and that this is conservative when compared to the actual computed stresses of less than 2000 psi. Based on the margins available between required strength and analyzed strength, we find this acceptable for restart. However, to provide long-term assurance that the concrete structure is capable of carrying its applicable loads, we will require that the licensee provide further confirmation of its evaluation by performing an analysis of the concrete structure assuming conservatively high temperature values and conservatively low concrete strength through the wall.

B. Equipment and Components

Our evaluation of the individual components identified as being subject to the high drywell temperatures experienced is as follows:

1. ASCO solenoid valve, model no. NP8320A184E

Initial qualification testing of these valves was performed in a 268F ambient temperature with the solenoids in the energized condition. The manufacturer's catalog data indicate that energization of the solenoids results in a 144F temperature rise above ambient. Using this assumption, a qualified life of several thousand years would be expected for a deenergized valve (the normal operating mode) in a 190F environment. However, even if credit is only given for about one-half of the solenoid temperature rise as a result of energization, the calculated remaining qualified life of the ASCO solenoids is more than one operating cycle. Consequently, we conclude that continued operation is justified.

2. AVCO solenoid valves, model no. C5159

No aging test was performed on the subject valves but the licensee has provided a detailed aging analysis based on the Arrhenius methodology. Based on this analysis the licensee has established that the qualified life of the subject valves is eight years. However, based on more recent information from the licensee on March 19, 1982, the licensee stated that all nonmetallic parts of the subject valves have been replaced. Based on this replacement, we find that continued operation is justified.

3. NAMCO limit switches, model no. EA740-50100

The licensee has performed the aging analysis and based on the analysis has replaced the Buna-N gaskets and EPR seals with new silicone rubber material for all affected limit switches. The licensee has also indicated that both grease and oil have a manufacturer's rating of 400F to 500F and therefore do not require replacement. Based on our evaluation of the licensee's assessment, we find that continued operation is justified.

4. Target rock solenoid valves, model no. 1/2 SMS-A-01

The licensee has replaced the subject valves with new Target Rock solenoid valves that are qualified to IEEE-323, 1974 and have a six-year manufacturer's recommended maintenance interval. Based on this, we find that continued operation is justified.

5. Limitorque valve actuators, model SMB-(various)

The licensee has performed an aging analysis and visual inspection and based on that has verified that (a) Viton seals are used in all MOV's, (b) melamine or fiberite has been used for the limit switch material, (c) jumper wires for MO1001-63, MO1001-50 and MO1201-2 have been replaced, (d) all lubricants have been replaced, and (e) new limit switch gear frames and limit switch compartment covers are on order and will be replaced next refueling outage. Based on the above and since (1) Limitorque has judged the limit switch compartment cover acceptable for BWR operation and (2) no visual corrosion effects were found on the gear frames, we find that continued operation is justified.

6. TEC valve flow monitor system, model no. 1414

The licensee has indicated that the charge converter and cable assembly are new units and the only component of this system which has been exposed to high drywell temperature is the accelerometer sensor. The licensee has indicated that the sensor does not contain any age sensitive material and the connector is manufactured from a silicate compound with an expected life of 147,548 years. Based on our evaluation of the licensee's assessment, we find that continued operation is justified.

7. Junction boxes

The licensee has performed the aging analysis and based on this analysis has replaced gasket material for all junction boxes above the 30' elevation with the new silicone rubber gaskets. Gaskets below 30' elevation have a remaining expected life of 13.25 years at 152°F or qualified life of 4.4 years at 152°F. Based on our evaluation of the licensee's assessment, we find that continued operation is justified.

8. Ring tongue type termination

The use of sound installation practices to attach terminations to a qualified barrier-type terminal block assures that adequate clearance is provided so that failure of the lug shank insulation would not impair circuit operability. The licensee has either installed new, qualified ring tongue type terminations or, by inspection, verified proper installation and that adequate clearance between connections has been provided. Based on these considerations we find that continued operation is justified.

9. Kerite 600V power and control cable type FR/FR

The licensee has taken two samples, one from the 41' elevation and another one from the 73' elevation, and performed physical (elongation) and electrical tests. Based on these tests the licensee has listed the percentage elongation for sample 1 as 235 for the insulation and 250 for the jacket while for sample 2 the percentage elongation was 110 and 160, respectively. The licensee has also quoted the percentage elongation of 150 and 190 for the material aged to 40 years at 90C operating temperature. Based on these numbers it is apparent that sample 2 has lost more elongation than for the original test. However, it should be noted that the original testing for radiation was tested for 200 Mrads while the expected post-accident LOCA dose is only 64 Mrad. This indicates that the elongation measured after 40 years thermal aging and radiation will be less at Pilgrim than that predicted from the original test. Assuming the linear degradation in elongation and accounting for normal radiation dose, the licensee has evaluated that 4 1/2 years of remaining life is left for the subject cables between elevation 41' and 73'. Based on the above the staff agrees with the licensee's assessment that continued operation is justified.

10. Okonite power and control cable, Okonite insulation, Okoprene jacket

The licensee has replaced the subject cable above 41' elevation. For the cables below 41' elevation, the licensee has provided the analysis which indicated that the jacket might be damaged but the insulation still has the remaining qualified 6.2 years at the rated temperature of 90°C. The licensee has also indicated that the okonite cable at PNPS was qualified without the jacket. Based on the above and the fact that the drywell is inerted during power operation and the jacket is used for the purpose of flame retardancy, we agree with the licensee's assessment and find that continued operation is justified.

11. GE switchboard wire, type SIS

The licensee has performed the aging analysis which indicated that the qualified life at 160°F is 45.5 years while at the rated condition the cable is good for only 6.1 years. The licensee has also performed the equivalent degradation time the cable has been used and indicated the degradation time of 1.7 years at 160°F and 1.94 years at 194°F. Since the cable has been used since 1972, ten years of the life has been used at the rated temperature. However, since the cable will not operate at the rated condition and is expected to operate somewhere in between 160°F and 194°, we agree with the licensee's assessment and find that continued operation is justified. However, the staff also requires that the licensee should submit the new analysis based on the temperature rise based on cable use and demonstrate that continued operation is justified beyond one refueling outage.

12. Raychem cable splices, model WCSF-N

The licensee has performed an aging analysis and determined that the material which comprises this equipment is insensitive to thermal degradation for the range of temperatures to which it was exposed. We have reviewed the information provided by the licensee and find that continued operation is justified.

13. GE electrical penetration, cannister type

The licensee has performed an aging analysis and demonstrated that continued operation is justified. The staff agrees with the licensee's assessment except for the cables. The licensee is using the GE SIS type wires. The staff's position regarding these wires is covered in Item 11. Based on our evaluation of the licensee's assessment, we find that continued operation is justified.

14. Physical science electrical penetration

The licensee has performed an aging analysis and determined that this equipment contains no age sensitive materials which would compromise pressure boundary integrity. In addition, testing has been performed during this outage to establish the pressure integrity of this equipment. Based upon our review of the information provided by the licensee, we find that continued operation is justified.

15. Bergen Patterson snubbers

The licensee has performed an aging analysis and based on this analysis has replaced all snubbers above 44' elevation. All snubbers below 44' elevation have a remaining life of 6.9 years at 194°F. Based on our evaluation of the licensee's assessment, we find that continued operation is justified.

16. ITT Hammel-Dahl air operators

The licensee has performed an aging analysis and determined that Buna-N o-rings and Buna-N/nylon diaphragms have an expected life of only 39 days. The licensee has also stated that these air operators are fail safe and failure of any nonmetallic part will not impair the safe operation of the plant. The licensee will disassemble, inspect, and rebuild all air operators. Air regulators will be replaced. Based on our evaluation of the licensee's assessment, we find that continued operation is justified.

17. Hydroline air operators

The licensee has performed an aging analysis and determined that continued operations is justified. Based upon our review of the information provided by the licensee, we find that continued operations is justified.

C. Accident Analyses

The effect of high initial drywell temperature on design basis loss of coolant accident (LOCA) analyses was reviewed. The design basis LOCA results in the most severe drywell pressurization rate and peak pressure loading. Therefore, it is bounding for other accidents. The containment pressure/temperature response results for a number of higher initial drywell temperatures was compared with the results obtained from analyses assuming an initial drywell temperature of 135°F (average design drywell temperature). These analyses indicated that lower peak pressures and lower drywell pressurization rates resulted when higher initial drywell temperatures were assumed and that the effect on peak post-LOCA drywell temperatures was negligible.

These results are expected since higher initial drywell temperatures result in a lower air density and mass than at lower temperatures. Thus, both peak post-LOCA drywell pressure and drywell pressurization result. Likewise, the effect on peak temperatures would also be expected to be negligible because of the small (relative to post-LOCA heat addition) additional heat content of the drywell atmosphere and structure as a result of high initial drywell temperatures.

Finally, regression analysis results obtained from the Mark I Containment Program 1/4 Scale Test Program have demonstrated that torus pool swell loads (both downforce and upforce) will be lower for a higher initial drywell temperature.

Therefore, we conclude that operation with a drywell ambient temperature higher than the nominal design value does not adversely affect accident analyses previously performed.

D. Technical Specifications

The licensee has proposed Technical Specifications which provide LCOs and surveillance requirements for drywell temperatures and for drywell temperature monitoring instrumentation. These TS provide a drywell temperature limit of 194°F above elevation 40' and 150°F at or below elevation 40'. These temperatures were developed taking into consideration the long-term effects of ambient temperature on equipment design limits and materials of components required for accident mitigation or plant shutdown.

Upon exceeding the proposed TS temperature limits, an engineering evaluation is required to be performed to assess potential damage and render a determination as to the ability of safety related equipment to perform its intended functions.

In addition, if the drywell temperature at any elevation exceeds 215°F for more than 30 minutes, the proposed TS requires the plant to be in a cold shutdown condition within 24 hours.

Finally, limiting conditions for operation and surveillance requirements for drywell temperature monitoring instrumentation have been proposed which provide assurance that the temperature monitoring instrumentation is operable at different elevations throughout the drywell.

We have reviewed the proposed TS and have determined that they limit containment drywell temperatures to values which will not have an adverse impact on drywell equipment, components, and structures required for safe plant operation. Consequently, we find the proposed TS changes acceptable.

4.0 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 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.

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: March 20, 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. 58 to Facility Operating License No. DPR-35 issued to Boston Edison Company (the licensee) which revised the Technical Specifications for operation of the Pilgrim Nuclear Power Station (the facility) located near Plymouth, Massachusetts. The amendment is effective as of its date of issuance.

The amendment revises the Technical Specifications to incorporate limiting conditions for operation and surveillance requirements for drywell temperature and drywell temperature monitoring instrumentation.

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 it 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 the issuance of the amendment.

For further details with respect to this action, see (1) the application for amendment dated March 1, 1982, (2) Amendment No. 58 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, N.W., Washington, D.C., and at the Plymouth Public Library on North Street in Plymouth, Massachusetts 02360. A single 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 20th day of March 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

D B Vassallo

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