6/21/78

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Docket No. 50-293

Boston Edison Company
M/C Nuclear
ATTN: Mr. G. Carl Andognini
800 Boylston Street
Boston, Massachusetts 02199

Gentlemen:

In response to your request for license amendment dated December 1, 1976 and a supplement thereto dated February 23, 1977, the Commission has issued the enclosed Amendment No. 31 to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station Unit No. 1.

This amendment incorporates provisions into the facility Technical Specifications which establish limiting conditions for operation and surveillance requirements for drywell to suppression chamber differential pressure control and suppression pool water level.

These requirements provide assurance that facility operation will be in accordance with the assumptions utilized in your facility's plant-unique analysis which was performed in conjunction with the Mark I Containment Short Term Program evaluation.

The enclosed license amendment reflects those changes to your original request for license amendment which have been agreed to in discussions with your staff. These changes have been made to provide consistent requirements for all Mark I containment facilities.

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

Sincerely,

/s/

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Cont I

Enclosures:
1. Amendment No. 31 to DPR-35
2. Safety Evaluation
3. Notice

*SEE PREVIOUS YELLOW FOR CONCURRENCE

OFFICE➤	ORB#3	ORB#3	ORB#3	PSYB		
SURNAME➤	*SSheppard	*JHannon:acr	*TIppolito	*CGrimes		
DATE➤	5/17/78	5/23/78	6/ 21 /78	6/12/78		

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Docket No. 50-293

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The enclosed license amendment reflects those changes to your original request for license amendment which have been agreed to in discussions with your staff. These changes have been made to provide consistent requirements for all Mark I containment facilities.

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

Sincerely,

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

pending resolution of # of ΔP channels OK

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

OFFICE		ORB#3	ORB#3	~~OELD~~	ORB#3	PSYB
SURNAME		SSheppard	JHannon:acr	NA	Ippolito ~~GLear~~	CGrimes
DATE		5/17/78	5/23/78	~~5/ /78~~	6/21/78	6/12/78

NRC FORM 318 (9-76) NRCM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

cc w/enclosures:
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JFK Federal Building
Boston, Massachusetts 02203

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Washington, D. C. 20005

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Plymouth, Massachusetts 02360

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Water Quality & Environmental Commissioner
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Boston, Massachusetts 02202

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Chairman, Board of Selectmen
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Chief, Energy Systems Analyses
Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
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401 M Street, S. W.
Washington, D. C. 20460

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

BOSTON EDISON COMPANY

DOCKET NO. 50-293

PILGRIM NUCLEAR POWER STATION UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 31
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 December 1, 1976 and a supplement thereto dated February 23, 1977, comply 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:

   3.B <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 31, 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

T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 21, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 31

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 vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

57/58
65/66
152A
171/172

Add page 152B.

PNPS

TABLE 3.2.E

INSTRUMENTATION THAT MONITORS DRYWELL LEAK DETECTION

Minimum # of Operable Instrument Channels	Instrument	Action
1	Equipment Drain Sump Flow Integrator	A
1	Floor Drain Sump Flow Integrator	A
1	Air Sampling System	A

NOTES FOR TABLE 3.2.E

1. The two (2) flow integrators, one for the equipment drain sump and the other for the floor drain sump, comprise the basic instrument system.

   An alternate system to determine the leakage flow is a manual system whereby the time between sump pump starts is monitored. This time interval will determine the leakage flow because the volume of the sump will be known.

2. Action

A. Whenever the reactor coolant leakage system is required to be operable, there shall be one operable system, or the reactor shall be placed in a Cold Shutdown Condition within 24 hours. **Refer to Specification 3.6.C.**

PNPS

TABLE 3.2.F

SURVEILLANCE INSTRUMENTATION

Minimum # of Operable Instrument Channels	Instrument #	Instrument	Type Indication and Range	Notes
2	640-29A & B	Reactor Water Level	Indicator 0-60"	(1) (2) (3)
2	640-25A & B	Reactor Pressure	Indicator 0-1200 psig	(1) (2) (3)
2	TRU-9044 TRU-9045	Drywell Pressure	Recorder 0-80 psia	(1) (2) (3)
2	TRU-9044 TI- 9019	Drywell Temperature	Recorder, Indicator 0-400°F	(1) (2) (3)
2	TRU-9045 TI- 9018	Suppression Chamber Air Temperature	Recorder, Indicator 0-400°F	(1) (2) (3)
2	LR- 5038 LR- 5049	Suppression Chamber Water Level	Recorder 0-32"	(1) (2) (3)
1	NA	Control Rod Position	28 Volt Indicating Lights	(1) (2) (3) (4)
1	NA	Neutron Monitoring	SRM, IRM, LPRM 0 to 100% power	
2	TI- 5047 TI- 5048	Suppression Chamber Water Temperature	Indicator 50-150°F	(1) (2) (3)
1	PI-5021	Drywell/Torus Diff. Pressure	Indicator -.25→3.0 psid	(1) (2) (3) (4)
1 1	PI-5067A	Drywell Pressure	Indicator -.25→3.0 psig	(1) (2) (3) (4)
1	PI-5067B	Torus Pressure	Indicator -1.0→+2.0 psig	

PNPS

TABLE 4.2.E

MINIMUM TEST AND CALIBRATION FREQUENCY FOR DRYWELL LEAK DETECTION

Instrument Channel	Instrument Functional Test	Calibration Frequency	Instrument Check
1) Equipment Drain Sump Flow Integrator	(1)	Once/3 months	Once/day
2) Floor Drain Sump Flow Integrator	(1)	Once/3 months	Once/day
3) Air Sampling System	(1)	Once/**3 months**	Once/day

Amendment No. 31

PNPS

TABLE 4.2.F

MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION

	Instrument Channel	Calibration Frequency	Instrument Check
1)	Reactor Level	Once/6 months	Each Shift
2)	Reactor Pressure	Once/6 months	Each Shift
3)	Drywell Pressure	Once/6 months	Each Shift
4)	Drywell Temperature	Once/6 months	Each Shift
5)	Suppression Chamber Temperature	Once/6 months	Each Shift
6)	Suppression Chamber Water Level	Once/6 months	Each Shift
7)	Control Rod Position	NA	Each Shift
8)	Neutron Monitoring	(2)	Each Shift
9)	Drywell/Torus Differential Pressure	Once/6 months	Each Shift
10)	Drywell Pressure	Once/6 months	Each Shift
10) 11)	Torus Pressure	Once/6 months	

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 1.50 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 1.50 psid 24 hours prior to a scheduled shutdown.

k. The differential pressure may be reduced to less than 1.50 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 to provide downcomer submergence of between 3.75 and 4.00 feet.

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.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing "open vessel" physics tests at power levels not to exceed 5 Mw(t).	2. Integrated Leak Rate Testing

2. Integrated Leak Rate Testing

a. The primary containment integrity shall be demonstrated by performing an Integrated Primary Containment Leak Test (IPCLT) in accordance with either Method A or Method B, as follows:

Method A

Perform leak rate test prior to initial unit operation at the test pressure of 45 psig, P_t (45), to obtain measured leak rate L_m (45), or

Method B

Perform leak rate test prior to initial unit operation at the test pressure of 45 psig, P_t (45), and 23 psig, P_t (23), to obtain the measured leak rates, L_m (45) and L_m (23), respectively.

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. In conjunction with the Mark I Containment Short Term Program, a plant unique analysis(1) was performed which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping. The maintenance of a drywell-suppression chamber differential pressure of 1.50 psid and a suppression chamber water level corresponding to a downcomer submergence range of 3.75 to 4.00 feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

(1) Plant Unique Analysis Report for Torus Support System and Attached Piping for Pilgrim Unit 1 Nuclear Power Station, Teledyne Technical Report No. TR 2255(a) dated August 5, 1976.

BASES:

3.7.B and 3.7.C

Standby Gas Treatment System and Secondary Containment

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation, when the drywell is sealed and in service; the reactor building provides primary containment when the reactor is shutdown and the drywell is open, as during refueling. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required as well as during refueling.

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the stack during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environs. One standby gas treatment fan is designed to automatically start upon containment isolation and to maintain the reactor building pressure at approximately a negative 1/4-inch water gauge pressure; all leakage should be in-leakage. Should the fan fail to start, the redundant alternate fan and filter system is designed to start automatically. If one standby gas treatment system train is inoperable, the other circuit must be tested daily. This substantiates the availability of the operable train and results in no added risk; thus, reactor operation or refueling operation can continue. If neither train is operable, the plant is brought to a condition where the system is not required.

While only a small amount of particulates is released from the pressure suppression chamber system as a result of the loss of coolant accident, high efficiency particulate filters before and after the charcoal filters are specified to minimize potential particulate release to the environment and to prevent clogging of iodine filters. The high-efficiency filters have an efficiency greater than 99% for particulate matter larger than 0.3 micron. The minimum iodine removal efficiency is 99%. Filter banks will be replaced whenever significant changes in filter efficiency occur. Tests of impregnated charcoal identical to that used in the filters indicate that shelf life up to five years leads to only minor decreases in methyl iodine removal efficiency.

The efficiency of 99% of the charcoal and particulate filters is sufficient to prevent exceeding 10CFR100 guidelines for the accidents analyzed. The analysis of the loss of coolant accident assumed a charcoal filter efficiency of 95%, and TID 14844 fission product source term. Hence, requiring 99% efficiency for both the charcoal and particulate filters provides adequate margin. A 14 kw heater maintains relative humidity below 70% in order to assure the efficient removal of methyl iodine on the impregnated charcoal filters.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 31 TO LICENSE NO. DPR-35

BOSTON EDISON COMPANY

PILGRIM NUCLEAR POWER STATION UNIT NO. 1

DOCKET NO. 50-293

Introduction

In conjunction with the Short Term Program (STP) evaluation of Boiling Water Reactor facilities with the Mark I containment system, the Boston Edison Company (the licensee) submitted a Plant Unique Analysis (PUA) for the Pilgrim Nuclear Power Station Unit No. 1. This analysis was performed to confirm the structural and functional capability of the containment suppression chamber and attached piping, to withstand newly-identified suppression pool hydrodynamic loading conditions which had not been explicitly considered in the original design analysis for the plant. As part of the STP evaluation, specific loading conditions were developed for each Mark I facility, to account for the change in the magnitude of the loads due to plant-specific variations from the reference plant design for which the basic loading conditions were developed.

The results of the NRC staff's review of the hydrodynamic load definition techniques and the Mark I containment plant unique analyses are described in the "Mark I Containment Short Term Program Safety Evaluation Report", NUREG-0408, December 1977. As discussed in this report, the NRC staff has concluded that each Mark I containment system would maintain its integrity and functional capability in the unlikely event of a design basis loss-of-coolant accident (LOCA) and, therefore, that licensed Mark I BWR facilities can continue to operate safely, without undue risk to the health and safety of the public, during an interim period of approximately two years, while a methodical, comprehensive Long Term Program is conducted.

As discussed in Section III.C of NUREG-0408, of all of the plant parameters that were considered in the development of the hydrodynamic loads for the STP, only two parameters are expected to vary during normal plant operation; these are (1) the drywell-wetwell differential pressure; and (2) the suppression chamber (torus) water level. Subsequent to the submittal of the PUA, the licensee was requested to submit proposed Technical Specifications which assure that the allowable range of these two parameters during facility operation would be in accordance with the values utilized in the PUA.

The licensee has been operating this facility with differential pressure control to enhance the safety margins of the containment structure since early 1976. This evaluation provides a more detailed basis for establishing the allowable range of drywell-wetwell differential pressure and torus water level, in order to quantify containment safety margins. This amendment incorporates these parameters into the Technical Specifications with the associated limiting conditions for operation and surveillance requirements.

By letters dated December 1, 1976 and February 23, 1977, the licensee proposed changes to the facility Technical Specifications to incorporate limiting conditions for operation and surveillance requirements for differential pressure control and torus water level. Our evaluation of these proposed changes follows.

Evaluation

The licensee has proposed certain Technical Specification requirements for the purpose of assuring that the normal plant operating conditions are within the envelope of conditions considered in their PUA. These Technical Specification changes establish (1) limiting condition for operation (LCOs) for drywell to torus differential pressure and torus water level, and (2) associated surveillance requirements. All other initial conditions utilized in the PUA are either presently included in the Technical Specifications or are configurational conditions which have been confirmed by the licensee and will not change during normal operation.

Differential pressure between the drywell and the suppression chamber will result in leakage of the drywell atmosphere to the lower pressure regions of the reactor building and to the torus airspace. This leakage from the drywell will cause a slow decay in the differential pressure. Therefore, surveillance requirements for the differential pressure have been included in the Technical Specifications. Surveillance frequency of once per operating shift for the differential pressure was selected on the basis of previous operating experience.

The torus water level is not expected to vary significantly during normal operation, unless certain systems connected to the suppression pool are activated. The torus water level would normally be monitored whenever such systems are in use. Therefore, we find that inclusion of periodic torus water level surveillance requirements in the Technical Specifications is not required.

We have reviewed the differential pressure and torus water level monitoring instrumentation systems proposed by the licensee with regard to the number of available channels and the instrumentation accuracy. This type of instrumentation is typically calibrated at six-month intervals. To assure proper operation during such intervals, two monitoring channels for both differential pressure and torus water level have been provided, such that a comparison of the readings will indicate when one of the channels is inoperative or drifting. The errors in the instrumentation are sufficiently small relative to the magnitude of the measurement (i.e., a maximum differential pressure measurement error of 0.1 psid in a measurement of 1.0 to 2.0 psid and a maximum torus water level measurement error of 10% of the difference between the maximum and minimum torus water level) that they may be neglected, based on the expected load variation with differential pressure and torus water level.

There are certain periods during normal plant operations when the differential pressure control cannot be maintained. Therefore, provisions have been included in the Technical Specifications to relax the differential pressure/control requirements during specified periods. The justification for relaxing the differential pressure control during these specific periods and the basis for selecting the duration of the periods are discussed in detail below.

1. <u>Startup and Shutdown</u>

    During plant startup and shutdown, the drywell atmosphere undergoes significant barometric changes due to the variation in heat loads from the primary and auxiliary systems. In addition, it is during these periods that the drywell is being either inerted with nitrogen gas or deinerted. In order to keep the periods during which the differential pressure control is not fully effective as short as is reasonable, we have limited the relaxation of the differential pressure control requirements for the startup and shutdown periods to 24 hours following startup and 24 hours prior to a shutdown. This time period was selected on a basis similar to that for the inerting requirements, already existing in the Technical Specifications. The postulated design basis accident for the containment assumes that the primary system is at operating pressure and temperature. During the startup and shutdown transients, the primary system is at operating pressure and temperature for only a part of the transient, during which the differential pressure is being established. These time periods have been shown by previous operating experience to be adequate with respect to the startup and shutdown transients and at the same time sufficiently small in comparison to the duration of the average power run. Since the principal accident event to which differential pressure control is important to assure containment integrity (i.e., with a factor of safety of two) is a large break LOCA, we have considered whether there is a significantly greater probability of a large break LOCA during the startup and shutdown transients. We have concluded that there is not. Further, the operation of the plant systems is monitored more closely than normal during these periods and a finite magnitude of differential pressure will be available during the majority of these periods to mitigate the potential consequences of an accident.

2. Testing and Maintenance

During normal operation, there are a number of tests which are required to be conducted to demonstrate the continued functional performance of engineered safety features. The testing of certain systems will require, or result in, a reduction in the drywell-torus differential pressure. The operability testing of the drywell-torus vacuum breakers requires the removal of the differential pressure to permit the vacuum breakers to open. For the testing of high-energy systems (e.g., high pressure coolant injection pumps) during normal operation, the discharge flow is routed to the suppression pool. This energy deposition will raise the temperature of the suppression pool, resulting in an increase in torus pressure and a reduction in the differential pressure.

Functional performance testing of engineered safety features is necessary to assure proper maintenance of these systems throughout the life of the plant. Some of these tests (i.e, pump operability and drywell-wetwell vacuum breakers) may require or result in a reduction in the differential pressure. We estimate that not more than four tests will be required each month which will result in a reduction in differential pressure. In order to keep the periods during which the differential pressure control is not fully effective as short as is reasonable, we have permitted a relaxation of differential pressure control in order to conduct these tests, limited to a period of up to four hours. Again, we have carefully considered whether the probability of a large LOCA is significantly greater during these testing periods than that during normal operation. We conclude that it is not. Moreover, only the test of the drywell-wetwell vacuum breakers requires complete removal of the differential pressure.

Provisions have also been included in the Technical Specifications for performing maintenance activities on the differential pressure control system and for resolving operational difficulties which may result in an inadvertent reduction in the differential pressure for a short period of time. In certain circumstances, corrective action can be taken without having to attain a cold shutdown condition. To avoid repeated and unnecessary partial cooldown cycles, a restoration period has been incorporated into the action requirements of the LCO for differential pressure control; i.e., in the event that the differential pressure cannot be restored in six hours, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours. The six hour restoration period was selected on the basis that it represents an adequate minimum period of time during which any short-term malfunctions could be corrected, coupled with the minimum period of time required to conduct a controlled shutdown. The allowable time to conduct a controlled shutdown has been minimized, because the containment transient response is more a function of the primary system pressure than the reactor power level. On this basis, we find the proposed restoration period and action requirement acceptable.

We conclude that the limits imposed on the periods of time during which operation is permitted without the differential pressure control fully effective provides adequate assurance of overall containment integrity, and the periods of time differential pressure control is completely removed are acceptably small.

Environmental Consideration

We have determined that the amendment does not authorize 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 this amendment.

Conclusions

The proposed Technical Specifications will provide the necessary assurance that the plant's operating conditions remain within the envelope of the conditions assumed in the Plant Unique Analysis (PUA) performed in conjunction with the Mark I Containment Short Term Program. The PUA supplements the facility's Final Safety Analysis Report (FSAR) in that it demonstrates the plant's capability to withstand the suppression pool hydrodynamic loads which were not explicitly considered in the FSAR. We therefore conclude that the proposed changes to the Technical Specifications are acceptable.

We further conclude, 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: June 21, 1978

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. 31 to Facility Operating License No. DPR-35, issued to Boston Edison Company, which revised the Technical Specifications for operation of the Pilgrim Nuclear Power Station, Unit No. 1, located near Plymouth, Massachusetts. The amendment is effective as of the date of its issuance.

The amendment revised the Technical Specifications to incorporate requirements for establishing and maintaining the drywell to suppression chamber differential pressure and suppression chamber water level, to maintain the margins of safety established in the NRC staff's, "Mark I Containment Short Term Program Safety Evaluation", NUREG-0408. Operation in accordance with the conditions specified in NUREG-0408 has been previously authorized in 43 FR 13105, March 29, 1978.

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) application for amendment dated December 1, 1976 as supplemented February 23, 1977, (2) Amendment No. 31 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 Operating Reactors.

Dated at Bethesda, Maryland, this 21 day of June 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

TA Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors