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. 165
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 dated July 14, 1995, as supplemented September 12 and December 8, 1995, 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 amended by changes to the Technical Specifications as indicated in the attachment to this license amendment.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel G. McDonald Jr.

for Ledyard B. Marsh, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 23, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 165

FACILITY OPERATING LICENSE NO. DPR-35

DOCKET NO. 50-293

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
B2-2	B2-2
3/4.3-4	3/4.3-4
3/4.3-5	3/4.3-5
B3/4.3-6	B3/4.3-6
3/4.11-2	3/4.11-2
3/4.11-3	3/4.11-3

FUEL CLADDING INTEGRITY (2.1.1) (Cont)

Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28 x 10^3 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28 x 10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis, GETAB (1), which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the General Electric Critical Quality (X) - Boiling Length (L), GEXL, correlation.

The GEXL correlation is valid over the range of conditions used in the tests of the data used to develop the correlation. These conditions are:

Pressure:	800 to 1300 psia
Mass Flux:	0.1 to 1.5 Mlb/hr-ft^2
Inlet Subcooling:	0 to 70 Btu/lb
Axial Profile:	1.5 chopped cosine
	1.7 inlet peaked
	1.7 outlet peaked
R-Factor	0.95 to 1.20
Rod Array	9X9 GE 11 array

MINIMUM CRITICAL POWER RATIO (2.1.2)

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not result in damage to BWR fuel rods, the critical power at

(Cont)

LIMITING CONDITION FOR OPERATION

3.3 REACTIVITY CONTROL (Cont)

B. Control Rods (Cont)

4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second.

5. The RBM shall be operable as required in Table 3.2.C-1, or control rod withdrawal shall be blocked.

C. Scram Insertion Times

1. Average scram insertion time for all operable control rods from de-energization of the scram pilot valve solenoids to drop out (DO) of Notches 04, 24, 34, and 44 shall be no greater than:

Notch Position	Average Scram Times (seconds)
44 DO	0.508
34 DO	1.252
24 DO	2.016
04 DO	3.578

SURVEILLANCE REQUIREMENT

4.3 REACTIVITY CONTROL (Cont)

B. Control Rods (Cont)

4. Prior to control rod withdrawal for startup or during refueling, verify that at least two source range channels have an observed count rate of at least three counts per second.

C. Scram Insertion Times

1. Following each refueling outage, or after a reactor shutdown that is greater than 120 days, each operable control rod shall be subjected to scram time tests from the fully withdrawn position. If testing is not accomplished with the nuclear system pressure above 950 psig, the measured scram insertion time shall be extrapolated to reactor pressures above 950 psig using previously determined correlations. Testing of all operable control rods shall be completed prior to exceeding 40% rated thermal power.

LIMITING CONDITION FOR OPERATION

3.3 REACTIVITY CONTROL (Cont)

C. Scram Insertion Time (Cont)

2. Average scram insertion time for the three fastest operable control rods in each group of four control rods in all two-by-two arrays from de-energization of the scram pilot valve solenoids to dropout (DO) of Notches 04, 24, 34 and 44 shall be no greater than:

Notch Position	Average Scram Time (Seconds)
44 DO	0.538
34 DO	1.327
24 DO	2.137
04 DO	3.793

3. The maximum scram insertion time for any operable control rod from de-energization of the scram pilot valve solenoids to dropout of Notch 04 shall not exceed 7.00 seconds.

D. Control Rod Accumulators

At all reactor operating pressures, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around this rod has a:

1. Inoperable accumulator.
2. Directional control valve electrically disarmed while in a non-fully inserted position.
3. Scram insertion time greater than the maximum permissible insertion time.

If a control rod with an inoperable accumulator is inserted "full-in" and its directional control valves are electrically disarmed, it shall not be considered to have an inoperable accumulator.

SURVEILLANCE REQUIREMENT

4.3 REACTIVITY CONTROL (Cont)

C. Scram Insertion Time (Cont)

2. Within each 120 days of operation, a minimum of 10% of the control rod drives, on a rotating basis, shall be scram tested as in 4.3.C.1. An evaluation shall be completed every 120 days of operation to provide reasonable assurance that proper performance is being maintained.

D. Control Rod Accumulators

Once a shift, check the status of the pressure and level alarms for each accumulator.

BASES:

3/4.3 REACTIVITY CONTROL (Cont)

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the Safety Limit MCPR. Analysis of the limiting power transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above Specification, provide the required protection, and MCPR remains greater than the Safety Limit MCPR.

The scram times for all control rods will be determined at the time of each refueling outage. A representative sample of control rods will be scram tested during each cycle as a periodic check against deterioration of the control rod performance.

The limits on scram insertion time presented in Technical Specification 3.3.C include an allowance for the uncertainty in the location of the position indicator probes as well as an allowance for the uncertainty in the control rod positions themselves when dropout of the reed switches occur.

D. Control Rod Accumulators

Requiring no more than one inoperable accumulator in any nine-rod square array is based on a series of XY PDQ-4 quarter core calculations of a cold, clean core. The worst case in a nine-rod withdrawal sequence resulted in a k_{eff} <1.0 - other repeating rod sequences with more rods withdrawn resulted in k_{eff} >1.0. At reactor pressures in excess of 800 psig, even those control rods with inoperable accumulators will be able to meet required scram insertion times due to the action of reactor pressure. In addition, they may be normally inserted using the control-rod-drive hydraulic system. Procedural control will assure that control rods with inoperable accumulators will be spaced in a one-in-nine array rather than grouped together.

LIMITING CONDITIONS FOR OPERATION

3.11 REACTOR FUEL ASSEMBLY (Cont)

B. Linear Heat Generation Rate (LHGR)

During reactor power operation, the LHGR shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

C. Minimum Critical Power Ratio (MCPR)

1. During power operation MCPR shall be ≥ the MCPR operating limit specified in the Core Operating Limits Report. If at any time during operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

SURVEILLANCE REQUIREMENTS

4.11 REACTOR FUEL ASSEMBLY (Cont)

B. Linear Heat Generation Rate (LHGR)

The LHGR as a function of core height shall be checked daily during reactor operation at ≥ 25% rated thermal power.

C. Minimum Critical Power Ratio (MCPR)

1. MCPR shall be determined daily during reactor power operation at > 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.5.

2. The value of τ in Specification 3.11.C.2. shall be equal to 1.0 unless determined from the result of surveillance testing of Specification 4.3.C as follows:

   a) τ is defined as

$$\tau = \frac{\tau_{ave} - \tau_B}{1.252 - \tau_B}$$

LIMITING CONDITIONS FOR OPERATION

3.11 REACTOR FUEL ASSEMBLY (Cont)

C. Minimum Critical Power Ration MCPR (Cont'd)

2. The operating limit MCPR values as a function of the τ are given in Table 3.3.1 of the Core Operating Limits Report where τ is given by specification 4.11.C.2.

SURVEILLANCE REQUIREMENTS

4.11 REACTOR FUEL ASSEMBLY (Cont)

C. Minimum Critical Power Ration MCPR (Cont'd)

b. The average scram time to dropout of Notch 34 is determined as follows:

$$\tau_{ave} = \frac{\sum_{i=1}^{n} N_i \quad \tau_i}{\sum_{i=1}^{n} N_i}$$

Where: an n - number of surveillance tests performed to date in the cycle.

N_i - number of active control rods measured in the i[th] surveillance test.

τ_i - average scram time to dropout of Notch 34 of all rods measured in the i[th] surveillance test.

c. The adjusted analysis mean scram time (τ_B) is calculated as follows:

$$\tau_B = \mu + 1.65 \left[\frac{N_1}{\sum_{i=1}^{n} N_i} \right]^{1/2} \sigma$$

Where:

μ - mean of the distribution for average scram insertion time to dropout of Notch 34, 0.937 sec.

N_1 - total number of active control rods at BOC during the first surveillance test.

σ - standard deviation of the distribution for average scram insertion time to the dropout of Notch 34, 0.021 seconds.