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. 139
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 February 4, 1985, and revised on April 10, 1991 and June 13, 1991, 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 set forth in 10 CFR Chapter I;

   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:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 139, 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 its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Susan Shankman

Susan Shankman, Acting Project Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 29, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 139

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
5b	5b
44	44
57	57
65	65
72	72
125	125
- - -	125a*
- - -	125b*
126	126
143	143
144	144

*Denotes new page

1.0 DEFINITIONS (Continued)

Z. Offsite Dose Calculation Manual (ODCM) - An offsite dose calculation manual (ODCM) shall be a manual containing the current methodology and parameters to be used for the calculation of offsite doses due to radioactive gaseous and liquid effluents, the calculation of gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints, and the conduct of the Radiological Environmental Monitoring Program.

AA. Action - Action shall be that part of a specification which prescribes remedial measures required under designated conditions.

BB. Member(s) of the Public[1] - Member(s) of the public shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the site.

CC. Site Boundary[1] - The site boundary is shown in Figure 1.6-1 in the FSAR.

DD. Radwaste Treatment System

1. Gaseous Radwaste Treatment System - The gaseous radwaste treatment system is that system identified in Figure 4.8-2.

2. Liquid Radwaste Treatment System - The liquid radwaste treatment system is that system identified in Figure 4.8-1.

EE. Automatic Primary Containment Isolation Valves - Are primary containment isolation valves which receive an automatic primary containment group isolation signal.

FF. Pressure Boundary Leakage - Pressure boundary leakage shall be leakage through a non-isolable fault in a reactor coolant system component body, pipewall or vessel wall.

GG. Identified Leakage - Identified leakage shall be:

1. Reactor coolant leakage into drywell collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

2. Reactor coolant leakage into the drywell atmosphere from sources which are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be Pressure Boundary Leakage.

HH. Unidentified Leakage - Unidentified leakage shall be all reactor coolant leakage which is not Identified Leakage.

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[1] See FSAR Figure 1.6-1

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
E. Drywell Leak Detection	E. Drywell Leak Detection
The limiting conditions of operation for the instrumentation that monitors drywell leak detection are given in Section 3.6.C.	Instrumentation shall be functionally tested, calibrated and checked as indicated in Section 4.6.C.
F. Surveillance Information Readouts	F. Surveillance Information Readouts
The limiting conditions for the instrumentation that provide surveillance information readouts are given in Table 3.2.F.	Instrumentation shall be calibrated and checked as indicated in Table 4.2.F.

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3.2 BASES (Cont'd)

The flow comparator and scram discharge volume high level components have only one logic channel and are not required for safety.

The refueling interlocks also operate one logic channel, and are required for safety only when the mode switch is in the refueling position.

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Four radiation monitors are provided which initiate the Reactor Building Isolation and Control System and operation of the standby gas treatment system. The instrument channels monitor the radiation from the refueling area ventilation exhaust ducts.

Four instrument channels are arranged in a 1 out of 2 twice trip logic.

Trip settings of < 100 mr/hr for the monitors in the refueling area ventilation exhaust ducts are based upon initiating normal ventilation isolation and standby gas treatment system operation so that none of the activity released during the refueling accident leaves the Reactor Building via the normal ventilation path but rather all the activity is processed by the standby gas treatment system.

3.6.B Coolant Chemistry (Cont'd)

3. For reactor startups and for the first 24 hours after placing the reactor in the power operating condition, the following limits shall not be exceeded:

   Conductivity . . 10 μmho/cm
   Chloride ion . . 0.1 ppm

4. Except as specified in 3.6.B.3 above, the reactor coolant water shall not exceed the following limits when operating with steaming rates greater than or equal to 100,000 pounds per hour:

   Conductivity . . 10 μmho/cm
   Chloride ion . . 1.0 ppm

5. If Specification 3.6.B cannot be met, an orderly shutdown shall be initiated and the reactor shall be in Hot Shutdown within 24 hrs. and Cold Shutdown within the next 8 hours.

4.6.B Coolant Chemistry (Cont'd)

3. a. With steaming rates of 100,000 pounds per hour or greater, a reactor coolant sample shall be taken at least every 96 hours and analyzed for chloride ion content.

   b. When all continuous conductivity monitors are inoperable, a reactor coolant sample shall be taken at least daily and analyzed for conductivity and chloride ion content.

3.6.C Coolant Leakage

Any time irradiated fuel is in the reactor vessel and coolant temperature is above 212°F, the following limits shall be observed:

1. Operational Leakage

   a. Reactor coolant system leakage shall be limited to:

      1. No Pressure Boundary Leakage
      2. ≤5 gpm Unidentified Leakage
      3. ≤25 gpm Total Leakage averaged over any 24 hour period.

4.6.C Coolant Leakage

Any time irradiated fuel is in the reactor vessel and coolant temperature is above 212°F, the following surveillances shall be performed:

1. Operational Leakage

   Demonstrate drywell leakage is within the limits specified in 3.6.C.1 by monitoring the coolant leakage detection systems required to be operable by 3.6.C.2 at least once every 8 hours.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6.C.1 Operational Leakage (Cont'd) 4. $\leq$2 gpm increase in Unidentified Leakage within any 24 hour period when in RUN mode. b. With any reactor coolant system leakage greater than the limits of 2. and/or 3. above, reduce the leakage to within acceptable limits within 4 hours or be in at least Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours. c. With any reactor coolant system leakage greater than the limits of 4. above, identify the source of leakage within 4 hours or be in at least Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours. d. When any Pressure Boundary Leakage is detected be in at least Hot Shutdown within the next 12 hours and be in Cold Shutdown within the next 24 hours.	
2. Leakage Detection Systems a. The following reactor coolant system leakage detection systems shall be Operable: 1. One drywell sump monitoring system, and either	2. Leakage Detection Systems The following reactor coolant leakage detection systems shall be demonstrated Operable: a. For each required drywell sump monitoring system perform: 1. An instrument functional test at least once per 31 days, and

3.6.C.2 Leakage Detection Systems (Cont'd)

2. One channel of a drywell atmospheric particulate radioactivity monitoring system, or

3. One channel of a drywell atmospheric gaseous radioactivity monitoring system.

b. 1. At least one drywell sump monitoring system shall be Operable; otherwise, be in Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.

2. At least one gaseous or particulate radioactivity monitoring channel must be Operable; otherwise, reactor operation may continue for up to 31 days provided grab samples are obtained and analyzed every 24 hours, or be in Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.

c. With no required leakage detection systems Operable, be in Cold Shutdown within 24 hours.

4.6.C.2 Leakage Detection Systems (Cont'd)

2. An instrument channel calibration at least once per 18 months.

b. For each required drywell atmospheric radioactivity monitoring system perform:

1. An instrument check at least once per day,

2. An instrument functional test at least once per 31 days, and

3. An instrument channel calibration at least once per 18 months.

3.6.D. Safety and Relief Valves

1. During reactor power operating conditions and prior to reactor startup from a Cold Condition, or whenever reactor coolant pressure is greater than 104 psig and temperature greater than 340°F, both safety valves and the safety modes of all relief valves shall be operable. The nominal setpoint for the relief/safety valves shall be selected between 1095 and 1115 psig. All relief/safety valves shall be set at this nominal setpoint ± 11 psi. The safety valves shall be set at 1240 psig ± 13 psi.

2. If Specification 3.6.D.1 is not met, an orderly shutdown shall be initiated and the reactor coolant pressure shall be below 104 psig within 24 hours. Note: Technical Specifications 3.6.D.2 - 3.6.D.5 apply only when two Stage Target Rock SRVs are installed.

3. If the temperature of any safety relief discharge pipe exceeds 212°F during normal reactor power operation for a period of greater than 24 hours, an engineering evaluation shall be performed justifying continued operation for the corresponding temperature increases.

4.6.D. Safety and Relief Valves

1. At least one safety valve and two relief/safety valves shall be checked or replaced with bench checked valves once per operating cycle. All valves will be tested every two cycles.

2. At least one of the relief/safety valves shall be disassembled and inspected each refueling outage.

3. Whenever the safety relief valves are required to be operable, the discharge pipe temperature of each safety relief valve shall be logged daily.

4. Instrumentation shall be calibrated and checked as indicated in Table 4.2.F.

5. Notwithstanding the above, as a minimum, safety relief valves that have been in service shall be tested in the as-found condition during both Cycle 6 and Cycle 7.

BASES:

3.6.C and 4.6.C

Coolant Leakage

Allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes and on the ability to makeup coolant system leakage in the event of loss of offsite a-c power. The normally expected background leakage due to equipment design and the detection capability for determining coolant system leakage were also considered in establishing the limits. The behavior of cracks in piping systems has been experimentally and analytically investigated as part of the USAEC sponsored Reactor Primary Coolant System Rupture Study (the Pipe Rupture Study). Work utilizing the data obtained in this study indicates that leakage from a crack can be detected before the crack grows to a dangerous or critical size by mechanically or thermally induced cyclic loading, or stress corrosion cracking or some other mechanism characterized by gradual crack growth. This evidence suggests that for leakage somewhat greater than the limit specified for unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. However, the establishment of allowable unidentified leakage greater than that given in 3.6.C on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of 5 gpm, as specified in 3.6.C, the experimental and analytical data suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propogation. Leakage less than the magnitude specified can be detected reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time the plant should be shut down to allow further investigation and corrective action.

Verification of the integrity of the reactor coolant system (3.6.C.1.a.1.: No Pressure Boundary Leakage) is provided during RPV Class I system hydrostatic and leak tests conducted to meet section 3/4.6.G: Structural Integrity (ASME Code, Section XI, IWA 5000, and IWB 5000.)

Two leakage collection sumps are provided inside primary containment. Identified leakage is piped from pump seal leakoffs, reactor vessel head flange seal leakoff, selected valve stem leakoff including recirculation loop and main steam isolation valves, and other equipment drains to the drywell equipment drain sump. The second sump, the drywell floor drain collection sump receives leakage from the drywell coolers, control rod drives, other valve stems and flanges, floor drains, and closed cooling water system drains. Drainage into the drywell floor drain sump is generally considered Unidentified Leakage. Both sumps are equipped with level and flow monitoring equipment to alert operators if allowable leak rates are approached.

A drywell sump monitoring system, as required in 3.6.C.2, consists of one equipment sump pump and one floor drain sump pump, plus associated instrumentation. Flow integrators, one for the equipment drain sump and

BASES:

3.6.C and 4.6.C

Coolant Leakage (Continued)

another for the floor sump, comprise the basic instrument system, and are used to record the flow of liquid from the drywell sumps. A manual system whereby the time interval between sump pump starts is utilized to provide a back-up to the flow integrators if the instrumentation is found to be inoperable. This time interval determines the leakage flow because the capacity of the pump is known.

The capacity of each of the two drywell floor sump pumps is 50 gpm and the capacity of each of the two drywell equipment sump pumps is also 50 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

In addition to the sump monitoring of coolant leakage, airborne radioactivity levels of the drywell atmosphere is monitored by the Reactor Pressure Boundary Leak Detection System. This system consists of two panels capable of monitoring the primary containment atmosphere for particulate and gaseous radioactivity as a result of coolant leaks.

The 2 gpm limit for coolant leakage rate increase within any 24 hour period is a limit specified by the NRC in Generic Letter 88-01: "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping". This limit applies only during the RUN mode to accommodate the expected coolant leakage increase during pressurization.

The total leakage rate consists of all leakage, which flows to the drywell equipment drain sump (Identified leakage) and floor drain sump (Unidentified leakage).