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. 83
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 August 9, 1984, as amended by letters dated September 21, and October 19, 1984, 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 Appendix A, as revised through Amendment No. 83 , 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

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 7, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 83

FACILITY OPERATING LICENSE NO. DPR-35

DOCKET NO. 50-293

Remove	Insert
58	58
58a	58a
---	58b
59	59
66a	66a
66b	66b
---	66c
152	152
152a	152a
166	166

TABLE 3.2.F
SURVEILLANCE INSTRUMENTATION

Minimum # of Operable Instrument Channels	Instrument #	Parameter	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	(1) (2) (3) (4)
2	TI-5021-01A TRU-5021-01A	Suppression Chamber Water Temperature	Dual Indicator/ Multipoint Recorder 30-230°F (Bulk/Local)	(4) (1) (2) (3)
	TI-5022-01B TRU-5022-01B	Suppression Chamber Water Temperature	Dual Indicator/ Multipoint Recorder 30-230°F (Bulk/Local)	(4) (1) (2) (3)
1	PI-5021	Drywell/Torus Diff. Pressure	Indicator -.25 → 3.0 psig	(1) (2) (3) (4)
1	PI-5067A PI-5067B	Drywell Pressure Torus Pressure	Indicator -.25 → 3.0 psig Indicator -1.0 → +2.0 psig	(1) (2) (3) (4)

SURVEILLANCE INSTRUMENTATION

Minimum # of Operable Instrument Channels	Instrument #	Parameter	Type Indication and Range	Notes
1/Valve	a) Primary or b) Backup (5)	Safety/Relief Valve Position	a) Acoustic monitor b) Thermocouple	(5)
1/Valve	a) Primary or b) Backup (5)	Safety Valve Position Indicator	a) Acoustic monitor b) Thermocouple	(5)
1/Valve	See Note (6)	Tail Pipe Temperature Indication	Thermocouple	(6)
2	LI 1001-604A LR 1001-604A	Torus Water Level (Wide Range)	Indicator/Multipoint Recorder 0-300"H_2O	(4) (1) (2) (3)
	LI 1001-604B LR 1001-604B	Torus Water Level (Wide Range)	Indicator/Multipoint Recorder 0-300"H_2O	(4) (1) (2) (3)
2	PI 1001-600A PR 1001-600A	Containment Pressure, (High Range)	Indicator/Multipoint Recorder 0-225 psig	(4) (1) (2) (3)
	PI 1001-600B PR 1001-600B	Containment Pressure, (High Range)	Indicator/Multipoint Recorder 0-225 psig	(4) (1) (2) (3)
2	PI 1001-601A PR 1001-600A	Containment Pressure, (Low Range)	Indicator/Multipoint Recorder -5 to 5 psig	(4) (1) (2) (3)
	PI 1001-601B PR 1001-600B	Containment Pressure, (Low Range)	Indicator/Multipoint Recorder -5 to 5 psig	(4) (1) (2) (3)
2	RIT 1001-606A RIT 1001-606B RR 1001-606A RR 1001-606B	Containment High Radiation (Drywell)	Monitor/Multipoint Recorder 1 to 1x10' R/hr	(4) (7)

TABLE 3.2.F (Cont'd)
SURVEILLANCE INSTRUMENTATION

Minimum # of Operable Instrument Channels	Instrument #	Parameter	Type Indication and Range	Notes
1	RI 1001-607 RR 1001-608	Reactor Building Vent	Indicator/Multipoint Recorder 10^{-1} to 10^{4} R/hr	(4) (7)
1	RI 1001-608 RR 1001-608	Main Stack Vent	Indicator/Multipoint Recorder 10^{-1} to 10^{4} R/hr	(4) (7)
1	RI 1001-610 RR 1001-608	Turbine Building Vent	Indicator/Multipoint Recorder 10^{-1} to 10^{4} R/hr	(4) (7)

Notes for Table 3.2.F

(1) With less than the minimum number of instrument channels, restore the inoperable channel(s) within 30 days.

(2) With the instrument channel(s) providing no indication to the control room, restore the indication to the control room within seven days.

(3) If the requirements of notes (1) or (2) cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown Condition with 24 hours.

(4) These surveillance instruments are considered to be redundant to each other.

(5) At a minimum, the primary or back-up* parameter indicators shall be operable for each valve when the valves are required to be operable. With both primary and backup* instrument channels inoperable either return one (1) channel to operable status within 31 days or be in a shutdown mode within 24 hours.

The following instruments are associated with the safety/relief and safety valves:

Valve	Primary Acoustic Monitor	Secondary Tail Pipe Temperature Thermocouple
203-3A	ZT-203-3A	TE6271 *
203-3B	ZT-203-3B	TE6272 *
203-3C	ZT-203-3C	TE6273 *
203-3D	ZT-203-3D	TE6276 *
203-4A	ZT-203-4A	TE6274-B
203-4B	ZT-203-4B	TE6275-B

* See Note (6)

(6) At a minimum, for thermocouples providing SRV tail pipe temperature, one of the dual thermocouples will be operable for each SRV when the valves are required to be operable. If a thermocouple becomes inoperable, it shall be returned to an operable condition within 31 days or the reactor shall be placed in a shutdown mode within 24 hours.

(7) With less than the minimum number of operable instrument channels, restore the inoperable channels to operable status within 7 days or prepare and submit a special report to the Regional Director of Inspection and Enforcement within 14 days of the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the channels to operable status.

PNPS

TABLE 4.2.F (Cont.)

MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION

Instrument Channel	Calibration Frequency	Instrument Check
13) Torus Water Level (Wide Range)	Each refueling outage	Once every 30 days
14) Containment Pressure	Each refueling outage	Once every 30 days
15) Containment High Radiation	Each refueling outage	Once every 30 days
16) Reactor Building Vent Radiation Monitor	Each refueling outage	Once every 30 days
17) Main Stack Vent Radiation Monitor	Each refueling outage	Once every 30 days
18) Turbine Building Vent Radiation Monitor	Each refueling outage	Once every 30 days

PNPS

Table 4.2-G

Minimum Test and Calibration Frequency for
ATWS RPT/ARI Instrumentation

Instrument Channel	Instrument Functional Test (2)	Calibration (2)	Instrument Check (2)
1. Reactor High Pressure	(1)	Once/Operating Cycle-Transmitter	Once/day
		Once/3 months - Trip unit	Once/day
2. Reactor Low-Low Water Level	(1)	Once/Operating Cycle-Transmitter	Once/day
		Once/3 months - Trip unit	Once/day

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

3.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. At any time that the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water volume and temperature shall be maintained within the following limits except as specified in 3.7.A.2 and 3.7.A.3.

   a. Minimum water volume - 84,000 ft^3

   b. Maximum water volume - 94,000 ft^3

   c. Maximum suppression pool bulk temperature during normal continuous power operation shall be $\leq 80^{o}F$, except as specified in 3.7.A.1.e.

   d. Maxumum suppression pool bulk temperature during RCIC, HPCI or ADS operation shall be $\leq 90^{o}F$, except as specified in 3.7.A.1.e.

   e. In order to continue reactor power operation, the suppression chamber pool bulk temperature must be reduced to $\leq 80^{o}F$ within 24 hours.

   f. If the suppression pool bulk temperature exceeds the limits of Specification 3.7.A.1.d, RCIC, HPCI or ADS testing shall be terminated and suppression pool cooling shall be initiated.

   g. If the suppression pool bulk temperature during reactor power operation exceeds $110^{o}F$, the reactor shall be scrammed.

4.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment.

Specification:

A. Primary Containment

1. a. The suppression chamber water level and temperature shall be checked once per day.

   b. Whenever there is indication of relief valve operation or testing which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.

   c. Whenever there is indication of relief valve operation with the bulk temperature of the suppression pool reaching $160^{o}F$ or more and the primary coolant system pressure greater than 200 psig, an external visual examination of the suppression chamber shall be conducted before resuming power operation.

   d. Whenever there is indication of relief valve operation with the local temperature of the suppression pool T-quencher reaching $200^{o}F$ or more, an external visual examination of the suppression chamber shall be conducted before resuming power operation.

   e. A visual inspection of the suppresion chamber interior, including water line regions, shall be made at each major refueling outage.

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 bulk temperature reaches 120°F.

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

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

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

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

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

4.7 CONTAINMENT SYSTEMS (Cont'd)

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

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

BASES:

3.7.A & 4.7.A Primary Containment

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

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

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

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

Experimental data indicates that excessive steam condensing loads can be avoided if the peak local temperature of the pressure suppression pool is maintained below 200°F during any period of relief-valve operation with sonic conditions at the discharge exit. Analysis has been performed to verify that the local pool temperature will stay below 200°F and the bulk pool temperature will stay below 160°F for all SRV transients. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high pressure suppression chamber loadings.