May 28, 1997

Mr. E. Thomas Boulette, Ph.D
Senior Vice President - Nuclear
Boston Edison Company
Pilgrim Nuclear Power Station
RFD #1 Rocky Hill Road
Plymouth, MA 02360

SUBJECT: ISSUANCE OF AMENDMENT NO. 172 TO FACILITY OPERATING LICENSE NO. DPR-35, PILGRIM NUCLEAR POWER STATION (TAC NO. M97418)

Dear Mr. Boulette:

The Commission has issued the enclosed Amendment No. 172 to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station. This amendment is in response to your application dated November 26, 1996.

The amendment revises Technical Specifications (TSs) Definition 1.M, "Primary Containment Integrity," Note 6 on Table 3.2.A for the high flow main steam line instrumentation, Table 3.2.D for a typographical error, Table 3.2.F to reflect a change made in instrument type for the suppression chamber water temperature instrumentation, Table 3.2.F to reflect modifications made to suppression chamber bulk and local temperature instrumentation, Bases Section 3/4.6G to remove an obsolete reference to Group I welds, and Bases Section 3/4.7.A to remove "high radiation" from the description of Primary Containment Group 1 initiation signals. These changes correct typographical errors, add clarity and consistency, modify instrument numbers due to plant modifications and update the Bases. These changes are administrative in nature and have no effect on plant design, safety limit settings, or plant system operation. In addition, by letter dated March 7, 1997, changes were made to the Bases Section 3.10, "Core Alterations."

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register Notice.

Sincerely,

Original signed by

Alan B. Wang, Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-293

Enclosures: 1. Amendment No. 172 to License No. DPR-35
2. Safety Evaluation

cc w/encls: See next page

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AMENDMENT NO. 172 TO FACILITY OPERATING LICENSE NO. DPR-35-PILGRIM NUCLEAR POWER STATION

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cc: Plant Service list

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cc:

Mr. Leon J. Olivier
Vice President of Nuclear
Operations & Station Director
Pilgrim Nuclear Power Station
1FD #1 Rocky Hill Road
Plymouth, MA 02360

Resident Inspector
U. S. Nuclear Regulatory Commission
Pilgrim Nuclear Power Station
Post Office Box 867
Plymouth, MA 02360

Chairman, Board of Selectmen
11 Lincoln Street
Plymouth, MA 02360

Chairman, Duxbury Board of Selectmen
Town Hall
878 Tremont Street
Duxbury, MA 02332

Office of the Commissioner
Massachusetts Department of
Environmental Protection
One Winter Street
Boston, MA 02108

Office of the Attorney General
One Ashburton Place
20th Floor
Boston, MA 02108

Mr. Robert M. Hallisey, Director
Radiation Control Program
Massachusetts Department of
Public Health
305 South Street
Boston, MA 02130

Regional Administrator, Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Ms. Jane Fleming
8 Oceanwood Drive
Duxbury, MA 0233

Mr. Jeffery Keene
Licensing Division Manager
Boston Edison Company
600 Rocky Hill Road
Plymouth, MA 02360-5599

Ms. Nancy Desmond
Manager, Reg. Affairs Dept.
Pilgrim Nuclear Power Station
RFD #1 Rocky Hill Road
Plymouth, MA 02360

Mr. David F. Tarantino
Nuclear Information Manager
Pilgrim Nuclear Power Station
RFD #1, Rocky Hill Road
Plymouth, MA 02360

Ms. Kathleen M. O'Toole
Secretary of Public Safety
Executive Office of Public Safety
One Ashburton Place
Boston, MA 02108

Mr. Peter LaPorte, Director
Attn: James Muckerheide
Massachusetts Emergency Management
Agency
400 Worcester Road
P.O. Box 1496
Framingham, MA 01701-0317

Chairman, Citizens Urging
Responsible Energy
P.O. Box 2621
Duxbury, MA 02331

Citizens at Risk
P.O. Box 3803
Plymouth, MA 02361

W.S. Stowe, Esquire
Boston Edison Company
800 Boylston St., 36th Floor
Boston, MA 02199

cc: (cont.)

Chairman
Nuclear Matters Committee
Town Hall
11 Lincoln Street
Plymouth, MA 02360

Mr. William D. Meinert
Nuclear Engineer
Massachusetts Municipal Wholesale
Electric Company
P.O. Box 426
Ludlow, MA 01056-0426

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. 172
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 November 26, 1996, 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, and paragraph 2.C(2) of Facility Operating License No. DPR-35 is hereby amended to read as follows:

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

FOR THE NUCLEAR REGULATORY COMMISSION

Patrick D. Milano, Acting 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: May 28, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 72

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
1-3	1-3
3/4.2-9	3/4.2-9
3/4.2-24	3/4.2-24
3/4.2-26	3/4.2-26
B3/4.6-11	B3/4.6-11
B3/4.7-4	B3/4.7-4

L. Design Power - Design power means a steady-state power level of 1998 thermal megawatts.

M. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All manual containment isolation valves on lines connected to the reactor coolant system or containment which are not required to be open during accident conditions are closed.

2. At least one door in each airlock is closed and sealed.

3. All blind flanges and manways are closed.

4. All automatic primary containment isolation valves and all instrument line flow check valves are operable or at least one containment isolation valve in each line having an inoperable valve shall be deactivated in the isolated condition.

5. All containment isolation check valves are operable or at least one containment valve in each line having an inoperable valve is secured in the isolated position.

N. Secondary Containment Integrity - Secondary containment integrity means that the reactor building is intact and the following conditions are met:

1. At least one door in each access opening is closed.

2. The standby gas treatment system is operable.

3. All automatic ventilation system isolation valves are operable or secured in the isolated position.

O. Operating Cycle - Interval between the end of one refueling outage and the end of the next subsequent refueling outage.

P. Refueling Frequencies:

1. Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the plant after that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled outage; however, where such outages occur within 11 months of completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage. (Definitions U and V apply)

2. Refueling Interval - Refueling interval applies only to ASME Code, Section XI IWP and IWV surveillance tests. For the purpose of designating frequency of these code tests, a refueling interval shall mean at least once every 24 months.

NOTES FOR TABLE 3.2.A (Cont)

3. Instrument set point corresponds to 137.86 inches above top of active fuel.

4. Instrument set point corresponds to 79.86 inches above top of active fuel.

5. Not required in Run Mode (bypassed by Mode Switch).

6. Each steam line is monitored by two instrument trip channels per trip system. Therefore, the minimum number of main steam line high flow instruments required to be operable is four per main steam line unless the line is isolated.

7. These signals also start SBGTS and initiate secondary containment isolation.

8. Only required in Run Mode (interlocked with Mode Switch).

9. Deleted.

PNPS
TABLE 3.2.D

RADIATION MONITORING SYSTEMS THAT INITIATE AND/OR ISOLATE

Minimum # of Operable Instrument Channels Per Trip system (1)	Trip Function	Trip Level Setting	Action (2)
2	Refuel Area Exhaust Monitors	Upscale, <100 mr/hr	A or B
2	Refuel Area Exhaust Monitors	Downscale	A or B

NOTES FOR TABLE 3.2.D

1. Whenever the systems are required to be operable, there shall be two operable or tripped trip systems. If this cannot be met, the indicated action shall be taken.

2. Action

   A. Cease operation of the refueling equipment:
   B. Isolate secondary containment and start the standby gas treatment system.

TABLE 3.2.F (Cont)

SURVEILLANCE INSTRUMENTATION

Minimum # of Channels	Operable Instrument Instrument #	Parameter	Type Indication and Range	Notes
2	TI-5021-2A TRU-5021-1A	Suppression Chamber Water Temperature	Indicator/ Multipoint Recorder 30-230°F (Bulk)	(1) (2) (3) (4)
	TI-5022-2B TRU-5022-1B	Suppression Chamber Water Temperature	Indicator/ Multipoint Recorder 30-230°F (Bulk)	(1) (2) (3) (4)
1	PID-5021	Drywell/Torus Diff. Pressure	Indicator -.25 - +3.0 psig	(1) (2) (3) (4)
1	PID-5067A PID-5067B	Drywell Pressure Torus Pressure	Indicator -.25 - +3.0 psig Indicator - 1.0 - +2.0 psig	(1) (2) (3) (4)
1/Valve	(a)Primary or (b) Backup	Safety/Relief Valve Position	(a) Acoustic monitor (b) Thermocouple	(5)
1/Valve	(a)Primary or (b) Backup	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_20	(1) (2) (3) (4)
	LI-1001-604B LR-1001-604B	Torus Water Level (Wide Range)	Indicator/Multipoint Recorder 0-300" H_20	(1) (2) (3) (4)

G. Structural Integrity

The Pilgrim Nuclear Power Station Inservice Inspection Program conforms to the requirements of 10CFR50.55a(g). Where practical, the inspection of ASME Section XI Class 1, 2, and 3 components conforms to the edition and addenda of Section XI of the ASME Boiler and Pressure Vessel Code required by 10CFR50.55a(g). When implementation of an ASME Code required inspection has been determined to be impractical for PNPS, a request for relief from the inspection requirement is submitted to the NRC in accordance with 10CFR50.55a(g)(5)(iii).

Requests for relief from the ASME Code inspection requirements will be submitted to the NRC prior to the beginning of each 10 year inspection interval for which the inspection requirement is known to be impractical. Requests for relief from inspection requirements which are identified to be impractical during the course of the inspection interval will be reported to the NRC on an annual basis throughout the inspection interval.

I. Shock Suppressors (Snubbers)

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is therefore required that all snubbers required to protect the primary coolant system and all other safety related systems or components be operable during reactor operation.

The visual inspection frequency is based on maintaining a constant level of snubber protection to systems. The cumulative number of inoperable snubbers detected during any inspection interval is the basis for establishment of the subsequent inspection interval and the existing inspection interval should remain in effect until its completion.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable.

Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, and are exposed to the same environmental conditions such as temperature, radiation, and vibration.

When a snubber is found inoperable, an engineering evaluation is initiated, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. Initiating this evaluation within 72 hours ensures that prompt corrective action will be afforded.

A. Primary Containment (Cont)

capability of the structure over its service lifetime. Additional margin to maintain the containment in the "as built" condition is achieved by establishing the allowable operational leak rate. The allowable operational leak rate is derived by multiplying the maximum allowable leak rate or the allowable test leak rate by 0.75 thereby providing a 25% margin to allow for leakage deterioration which may occur during the period between leak rate tests.

The primary containment leak rate test frequency is based on maintaining adequate assurance that the leak rate remains within the specification. The leak rate test frequency is in accordance with 10CFR50 App. J.

The penetration and air purge piping leakage test frequency, along with the containment leak rate tests, is adequate to allow detection of leakage trends. Whenever a bolted double-gasketed penetration is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. It is expected that the majority of the leakage from valves, penetrations and seals would be into the reactor building. However, it is possible that leakage into other parts of the facility could occur. Such leakage paths that may affect significantly the consequences of accidents are to be minimized. The personnel air lock is tested at 10 psig, because the inboard door is not designed to shut in the opposite direction.

Primary Containment Isolation Valves

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss of coolant accident.

Group 1 - process lines are isolated by reactor vessel low-low water level in order to allow for removal of decay heat subsequent to a scram, yet isolate in time for proper operation of the core standby cooling systems. The valves in group 1 are also closed when process instrumentation detects excessive main steam line flow, low pressure, main steam space high temperature, or reactor vessel high water level.

Group 2 - isolation valves are closed by reactor vessel low water level or high drywell pressure. The group 2 isolation signal also "isolates" the reactor building and starts the standby gas treatment system. It is not desirable to actuate the group 2 isolation signal by a transient or spurious signal.

Group 3 - isolation valves can only be opened when the reactor is at low pressure and the core standby cooling systems are not required. Also, since the reactor vessel could potentially be drained through these process lines, these valves are closed by low water level.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 172 TO FACILITY OPERATING LICENSE NO. DPR-35

BOSTON EDISON COMPANY

PILGRIM NUCLEAR POWER STATION

DOCKET NO. 50-293

1.0 INTRODUCTION

By letter dated November 26, 1996, the Boston Edison Company (BECo or the licensee) submitted a request for changes to the Pilgrim Nuclear Power Station (PNPS) Technical Specifications (TSs). The requested changes would revise TS Definition 1.M, "Primary Containment Integrity," Note 6 on Table 3.2.A for the high flow main steam line instrumentation, Table 3.2.D for a typographical error, Table 3.2.F to reflect a change made in instrument type for the suppression chamber water temperature instrumentation, Table 3.2.F to reflect modifications made to suppression chamber bulk and local temperature instrumentation, Bases Section 3/4.6G to remove an obsolete reference to Group I welds, and Bases Section 3/4.7.A to remove "high radiation" from the description of Primary Containment Group 1 initiation signals. In addition, by letter dated March 7, 1997, changes were made to the Bases Section 3.10, "Core Alterations."

2.0 EVALUATION

The licensee requested a change to the PNPS TSs in accordance with 10 CFR 50.90. The proposed revisions are described and evaluated below:

(1) DEFINITION 1.M, "PRIMARY CONTAINMENT INTEGRITY"

Definition 1.M, "Primary Containment Integrity," will be modified to include the phrase "all instrument line flow check valves" to make this definition consistent with Limiting Condition for Operation (LCO) 3/4.7.A.2.a.4. The staff has reviewed the change and concurs that the addition of the phrase "all instrument line flow check valves" to the definition of primary containment integrity is consistent with LCO 3/4.7.A.2.a.4 and does not change the meaning of the definition. This change brings consistency between the definition and the LCO regarding primary containment integrity and is administrative in nature. Based on the above, the staff concludes that the proposed TS change is acceptable.

(2) NOTE 6 ON TABLE 3.2.A

Note 6 on Table 3.2.A for the high flow main steam line instrumentation that initiates primary containment isolation will be reworded for added clarity. The current wording of "Two [channels per trip system] required for each steam line" could be misinterpreted to mean a total of 8 instrument channels are needed. Actually, the four main steam lines have two instrument trip channels for each of the two trip systems. Therefore, 4 instruments per main steam line must be operable unless the line is isolated. Boston Edison had issued a memorandum to ensure proper understanding of this TS requirement. To further clarify this issue, BECo has proposed this change to reword the note in the TS. The staff has concluded that the revised note more clearly defines the plant configuration and, therefore, the proposed TS change is acceptable.

(3) TABLE 3.2.D

Table 3.2.D of the TS contains a typographical error. There are two notes labeled number 1. This change labels the second note as number 2. The staff concludes that this change is administrative in nature and, therefore, the proposed TS change is acceptable.

(4) TABLE 3.2.F

BECo has proposed to modify Table 3.2.F as a result of a change made in instrument type for the suppression chamber water temperature instrumentation. The instrument numbers and indication type have been changed for Instruments TI-5021-01A and TI 5022-01B. TI-5021-01A and TI 5022-01B have been physically changed and renumbered to TI-5021-02A and TI 5022-02B. Originally TI-5021-01A and TI-5022-01B were dual indicators. The original instruments indicated both suppression chamber bulk and local temperatures. A plant change replaced the dual instruments with individual instruments measuring bulk temperature only. This was acceptable as instrument requirements of TS 3.7.A.1.e through h only require bulk temperature. The TS Table will be modified to reflect that only bulk temperature is measured. TRU 5021-01A and TRU 5022-01B are multipoint recorders and have been renumbered to TRU 5021-1A and TRU 5022-1B. New instrument TI-5021-2A is the indicator for recorder TRU-5021-1A and new instrument TI-5022-2B is the indicator for recorder TRU-5022-1B. The staff has reviewed these changes and conclude they are administrative in nature and, therefore, the proposed change is acceptable.

The instruments for the drywell/torus differential pressure, drywell pressure and torus pressure instrumentation were changed. The nomenclature for these instruments will be changed to reflect this modification. The staff has reviewed these changes and conclude they are administrative in nature and, therefore, the proposed change is acceptable.

(5) BASES SECTION 3/4.6.G

An obsolete reference to Group I welds in Section 3/4.6G Bases will be removed. This Group I designation pre-dates the current ASME XI Code and should have been removed when the TSs were revised in Amendment No. 149, dated September 28, 1993. The ASME XI Code requirement is to assure high-stress welds are included in the weld selection process for inspection. The current PNPS Inservice Inspection Program includes this requirement; however, BECo no longer includes the Group I welds in the ISI Program unless they meet or exceed the code stress criteria. The Bases section has been updated to reflect the above.

(7) BASES SECTION 3/4.7.A

The description of Primary Containment Group 1 initiation signals in Bases 3/4.7.A is being revised to remove the words "high radiation." The main steam high radiation signal was removed from TSs in Amendment No. 154 dated June 21, 1994. This Bases reference change was inadvertently overlooked for that amendment and will be corrected by this submittal.

(8) BASES SECTION

By letter dated March 7, 1997, BECo informed the staff of a change to Bases Section 3.10, "Core Alterations." During last cycle, Pilgrim operated with some control rods completely inserted to suppress indications of a damaged fuel pin. During the current refueling outage, BECo removed the suspect fuel bundles early to begin inspection and reconstitution of the damaged bundle. Because a spiral unloading pattern of the fuel was planned, a change to the "Core Alterations" Bases was necessary to allow removal of the suspect fuel bundles out of the planned sequence. BECo has performed a 50.59 and concluded that removing fuel from five cells out of sequence is acceptable provided the fuel bundles removed do not completely surround and isolate a source range monitor from the rest of the core. This Bases Section will be modified to reflect the above.

The staff has concluded that these changes correct typographical errors, add clarity and consistency, modify instrument numbers due to plant modifications and update the Bases. These changes are administrative in nature and have no effect on plant design, safety limit settings, or plant system operation. The staff reviewed the request by BECo to revise the TS and Bases Section of the PNPS and based on the review, conclude that these revisions are administrative in nature and are acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Massachusetts State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (62 FR 6568). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: A. Wang

Date: May 28, 1997