November 10, 1998

Mr. Theodore A. Sullivan
Vice President Nuclear and Station Director
Boston Edison Company
Pilgrim Nuclear Power Station
RFD #1 Rocky Hill Road
Plymouth, MA 02360

SUBJECT: ISSUANCE OF AMENDMENT NO. 178 ,TO FACILITY OPERATING LICENSE NO. DPR-35, PILGRIM NUCLEAR POWER STATION (TAC NO. MA3443)

Dear Mr. Sullivan:

The Commission has issued the enclosed Amendment No. 178 to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station. This amendment is in response to your application dated June 26, 1998.

The amendment will modify various Pilgrim Nuclear Power Station Technical Specification pages to correct typographical errors, remove inadvertent replication of information, and update various Bases sections.

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.178 to License No. DPR-35
2. Safety Evaluation

cc w/encls: See next page

DFO1

DOCUMENT NAME: G:\PILGRIM\MA3443.AMD *SEE PREVIOUS CONCURRENCE

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DATED: November 10, 1998

AMENDMENT NO. 178 TO FACILITY OPERATING LICENSE NO. DPR-35 - PILGRIM NUCLEAR POWER STATION

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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. 178
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 June 26, 1998, 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, 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. 178 , 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

Cecil O. Thomas

Cecil O. Thomas, 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: November 10, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 178

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
3/4.1-2	3/4.1-2
3/4.2-20	3/4.2-20
3/4.3-3	3/4.3-3
B 3/4.7-2	B 3/4.7-2
B 3/4.7-4	B 3/4.7-4
B 3/4.7-5	B 3/4.7-5

<u>**PNPS Table 3.1.1 REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT**</u>

Operable Inst. Channels per Trip System (1)		Trip Function	Trip Level Setting	Modes in Which Function Must Be Operable			Action(1)
Minimum	Avail.			Refuel	Startup/Hot Standby	Run	
1	1	Mode Switch in Shutdown		X(7)	X		A
1	1	Manual Scram		X(7)	X	X	A
		IRM					
3	4	High Flux	≤ 120/125 of full scale	X(7)	X	(5)	A
3	4	Inoperative		X(7)	X	(5)	A
		APRM					
2	3	High Flux	(15)	(17)	(17)	X	A or B
2	3	Inoperative	(13)	X(7)	X(9)	X	A or B
2	3	High Flux (15%)	≤ 15% of Design Power	X(7)	X	(16)	A or B
2	2	High Reactor Pressure	≤ 1063.5 psig	X(10)	X	X	A
2	2	High Drywell Pressure	≤ 2.22 psig	X(8)	X(8)	X	A
2	2	Reactor Low Water Level	≥ 11.6 in. Indicated Level	X(10)	X	X	A
		SDIV High Water Level:	≤ 38 Gallons	X(2)(7)	X	X	A
2	2	East					
2	2	West					
4	4	Main Steam Line Isolation Valve Closure	≤ 10% Valve Closure	X(3)(6)	X(3)(6)	X(6)	A or C
2	2	Turbine Control Valve Fast Closure	≥ 150 psig Control Oil Pressure at Acceleration Relay	X(4)	X(4)	X(4)	A or D
4	4	Turbine Stop Valve Closure	≤ 10% Valve Closure	X(4)	X(4)	X(4)	A or D

TABLE 3.2.C.1 (Cont)

INSTRUMENTATION THAT INITIATES ROD BLOCKS

Trip Function	Operable Channels per Trip Function Minimum	Operable Channels per Trip Function Available	Required Operational Conditions	Notes
SRM Detector not in Startup Position	3	4	Startup/Refuel, except trip is bypassed when SRM count rate is ≥ 100 counts/second or IRMs on Range 3 or above	(1)(4)(6)
SRM Downscale	3	4	Startup/Refuel, except trip is bypassed when IRMs on Range 3 or above	(1)(4)(6)
SRM Upscale	3	4	Startup/Refuel, except trip is bypassed when the IRM range switches are on Range 8 or above (4)	(1)(4)(6)
SRM Inoperative	3	4	Startup/Refuel, except trip is bypassed when the IRM range switches are on Range 8 or above (4)	(1)(4)(6)
Scram Discharge Instrument Volume Water Level - High	2	2	Run/Startup/Refuel	(3)(6)
Scram Discharge Instrument Volume-Scram Trip Bypassed	1	1	Refuel/Shutdown	(3)(6)

LIMITING CONDITION FOR OPERATION

3.3 REACTIVITY CONTROL (CONT)

B. Control Rods (Cont)

2. The control rod drive housing support system shall be in place during reactor power operation and when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.

3. a. No control rods shall be moved when the reactor is below 20% rated power, except to shutdown the reactor, unless the Rod Worth Minimizer (RWM) is operable. A maximum of two rods may be moved below 20% design power when the RWM is inoperable if all other rods except those which cannot be moved with control rod drive pressure are fully inserted.

b. Control rod patterns and the sequence of withdrawal or insertion shall be established such that:

1) when the reactor is critical and below 20% design power the maximum worth of any insequence control rod which is not electrically disarmed is less than 0.010 delta k.

2) and when the reactor is above 20% design power the maximum worth of any control rod, including allowance for a single operator error, is less than 0.020 delta k.

SURVEILLANCE REQUIREMENT

4.3 REACTIVITY CONTROL (Cont)

B. Control Rods (Cont)

b. When the rod is fully withdrawn the first time subsequent to each refueling outage or after maintenance, observe that the drive does not go to the overtravel position.

2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.

3. Prior to control rod withdrawal for startup or insertion to reduce power below 20% of the operability of the Rod Worth Minimizer (RWM) shall be verified by:

a. verifying the correctness of the control rod withdrawal sequence input to the RWM computer.

b. performing the RWM computer diagnostic test.

c. verifying the annunciation of the selection errors of at least one out-of-sequence control rod in each distinct RWM group.

d. verifying the rod block function of an out-of-sequence control rod which is withdrawn no more than three notches.

BASES:

3/4.7 CONTAINMENT SYSTEMS (Cont)

A. Primary Containment (Cont)

The maximum permissible bulk suppression pool temperature of 120°F is acceptable since a complete accident blowdown can be accomodated without exceeding the bulk suppression pool temperature limit of 170°F immediately after blowdown. This 170°F LOCA blowdown limit is not a limit for the heatup of the suppression pool after the vessel is depressurized. 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. Current Technical Specification limits on suppression pool temperature ensure bulk pool temperature remains within an acceptable range to condense steam discharged to the suppression pool during a LOCA or SRV actuation.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and frequently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

If a loss-of-coolant accident were to occur when the reactor water temperature is below approximately 330°F, the containment pressure will not exceed the 62 psig code permissible pressure, even if no condensation were to occur. The maximum allowable pool temperature, whenever the reactor is above 212°F, shall be governed by this specification. Thus, specifying water volume-temperature requirements applicable for reactor-water temperature above 212°F provides additional margin above that available at 330°F.

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 leakage rate testing is based on the guidelines in Regulatory Guide 1.163 dated September 1995, NEI 94-01 Revision 0 dated July 25, 1995, and ANSI/ANS 56.8-1994. Specific acceptance criteria for as-found and as-left leakage rates, as well as methods of defining the leakage rates, are contained in the primary containment leakage rate testing program.

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, Option B and Regulatory Guide 1.163 dated September 1995.

Type A, Type B, and Type C tests will be performed using the technical methods and techniques specified in ANSI/ANS 56.8 - 1994 or other alternative testing methods approved by the NRC.

A note is included in Surveillance 4.7.A.2.a stating that definition 1.U is not applicable. The 25% allowable extension of surveillance intervals is already included in the primary containment leakage rate testing program; therefore, an additional 25% is not allowed.

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.

BASES:

3/4.7 CONTAINMENT SYSTEMS (Cont)

A. Primary Containment (Cont)

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.

Group 4 and 5 - process lines are designed to remain operable and mitigate the consequences of an accident which results in the isolation of other process lines. The signals which initiate isolation of group 4 and 5 process lines are therefore indicative of a condition which would render them inoperable.

Group 6 - process lines are normally in use and it is therefore not desirable to cause spurious isolation due to high drywell pressure resulting from non-safety related causes. To protect the reactor from a possible pipe break in the system, isolation is provided by high temperature in the cleanup system area or high flow through the inlet to the cleanup system. Also, since the vessel could potentially be drained through the cleanup system, a low level isolation is provided.

Group 7 - The HPCI vacuum breaker line is designed to remain operable when the HPCI system is required. The signals which initiate isolation of the HPCI vacuum breaker line are indicative of a break inside containment and reactor pressure below that at which HPCI can operate.

The maximum closure time for the automatic isolation valves of the primary containment and reactor vessel isolation control system have been selected in consideration of the design intent to prevent core uncovering following pipe breaks outside the primary containment and the need to contain released fission products following pipe breaks inside the primary containment.

In satisfying this design intent an additional margin has been included in specifying maximum closure times. This margin permits identification of degraded valve performance, prior to exceeding the design closure times.

In order to assure that the doses that may result from a steam line break do not exceed the 10CFR100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate that fuel rod cladding perforations would be avoided for main steam valve closure times, including instrument delay, as long as 10.5 seconds.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 178 TO FACILITY OPERATING LICENSE NO. DPR-35

BOSTON EDISON COMPANY

PILGRIM NUCLEAR POWER STATION

DOCKET NO. 50-293

1.0 INTRODUCTION

By letter dated June 26, 1998, the Boston Edison Company (BECo or the licensee) submitted a request to modify the Pilgrim Nuclear Power Station (PNPS) Technical Specifications (TS). The requested changes would make corrections to Table 3.1.1 item "Turbine Control Valve Fast Closure", and Table 3.2.C.1 item "Mode Switch in Shutdown". Section 3.3.B.3.b.1 will be changed as approved by the NRC in Amendment 39. Bases Section 3/4.7 will be revised to reflect new analysis and delete repetitive material.

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:

2.1 TABLE 3.1.1, "REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

For Table 3.1.1 item "Turbine Control Valve Fast Closure," a typographical error of "psit" was introduced in Amendment 169. It will be corrected to read "psig". The staff concludes that this change is administrative in nature and, therefore, the proposed TS change is acceptable.

2.2 TABLE 3.2.C.1, "INSTRUMENTATION THAT INITIATES ROD BLOCKS"

For Table 3.2.C.1 item "Mode Switch in Shutdown," a typographical error occurred when this item was added in Amendment 169. This item was incorrectly added to two pages of the table instead of only on one page. This item will be removed from page 3/4.2-20. In addition, the correct operable channels per trip function is two channels per trip function as stated on 3/4.2-21, not one channel per trip function as stated on page 3/4.2-20. The staff concludes that this change is administrative in nature and, therefore, the proposed TS change is acceptable.

2.3 SECTION 3.3.B.3.b.1, "REACTIVITY CONTROL"

Section 3.3.B.3.b.1 has been changed to read from "when the reactor is critical and below 10%..." to "when the reactor is critical and below 20%...". The licensee states that this change was approved by the NRC in Amendment 39 but was inadvertently not revised in the current copy of Technical Specifications. The staff has confirmed that Amendment 39 revised this number and concludes that this change is administrative in nature and, therefore, the proposed TS change is acceptable.

3.0 BASES SECTION

The licensee has revised BASES Section 3/4.7, "Containment Systems," to reflect changes approved by the NRC via Amendments 167, 172 and 173. In addition, Groups 1, 2 and 3 isolation signals are being deleted from page B 3/4.7-4 because identical information is provided on page B 3/4.7-5. The BASES sections will be changed to reflect the revised TSs and the changes are implemented and controlled by Regulatory Affairs Work Instruction 3.04-01, "Changes to Technical Specifications."

4.0 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.

5.0 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 (63 FR 50933). 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.

6.0 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: November 10, 1998