REGULATORY DOCKET FILE COPY

FEBRUARY 4 1980

Docket No. 50-293

Mr. G. Carl Andognini
Boston Edison Company
M/C NUCLEAR
800 Boylston Street
Boston, Massachusetts 02199

Dear Mr. Andognini:

The Commission has issued the enclosed Amendment No. 40 to Operating License No. DPR-35 for the Pilgrim Nuclear Power Station. This amendment consists of changes to the Technical Specifications in partial response to your requests dated July 6 and September 27, 1979. A related request dated December 27, 1977 was denied, as discussed below.

This amendment changes the Technical Specifications to (1) update Table 3.6.1 for safety related snubbers, (2) revise the station organization chart and the Operations Review Committee (ORC) composition, (3) reduce the response time requirement for ADS, (4) provide an alternate method of demonstrating reactivity margin, (5) add calibration and test frequencies to the IRM High Flux Instrument Channels, and (6) remove the power restriction requirement for testing MSIV closure times.

Also included in your submittals were requests to (1) change the method of calibrating the average power range monitoring (APRM) system, and (2) reduce the duration of the Integrated Primary Containment Leak Rate Test (IPCLRT) from 24 to 12 hours. These two requests cannot be approved at this time. The basis for our decision is detailed below.

The essential change in the method for calibrating the APRM system was to introduce a false gain characteristic to compensate for core peaking factors. We found that your submittal did not provide sufficient information to demonstrate how the revised APRM calibration procedure satisfies Paragraph 4.20 of IEEE Std. 279-1971. Specifically, the staff is concerned that the proposed practice of purposely introducing a false calibration may result in a condition where-in the operator will not be provided with accurate, complete, and timely information pertinent to APRM status and generating station safety. Therefore, this change cannot be supported by the staff.

The staff is presently in the process of developing criteria for reviewing the change concerning duration of IPCLRT. At this point, a schedule for developing the criteria is not available, due to the relatively low priority of this task. However, personnel are being assigned and a schedule will be developed. We will advise you of the schedule when it is available, and will provide you with the criteria when it is developed. A Technical Specification change will be issued when we have determined your submittal meets the review criteria.

OFFICE ▶					
SURNAME ▶					
DATE ▶					

Your letter dated December 27, 1977 requested a change to the Administrative Controls Section of the Technical Specifications to allow the individual performing the function of the Radiation Protection Manager to be located either on-site or off-site. We have reviewed your submittal and determined that your request cannot be approved, for the following reasons.

Regulatory Guide 8.8, Revision 2, Section C.1.b.(3) states that the Radiation Protection Manager (onsite) should have qualifications as presented in Regulatory Guide 1.8. The word "onsite" was purposefully added to the section on "Personnel Qualification and Training" of Regulatory Guide 8.8, Revision 1 so that the staff would have some basis for confidence that well trained and experienced radiation protection personnel (as described in Regulatory Guide 1.8) would be immediately available to onsite personnel to ensure that their exposures under normal and accident conditions were kept "as low as reasonably achievable" (ALARA). Although the qualifications of the RPM in Regulatory Guide 1.8 does not attest to his competency or expertise, it does provide a basis for assuring that the individual assigned that job responsibility is a knowledgeable professional who can provide the supervision necessary to implement the program required for radiation protection. The practical requirement for "onsite" RPM management is that radiation protection management, surveillance, and dose monitoring may be maintained by qualified health physics personnel during day-to-day plant operations. The onsite Radiation Protection Manager is also supported by onsite plant management. Together they must have a day-to-day working relationship in order to (1) implement the station ALARA program effectively, (2) resolve site specific questions relating to the ALARA program, (3) effectively handle emergency situations involving overexposed or contaminated site personnel, and (4) provide expertise in decision-making with respect to accidents that may have a potential for release of radioactive materials to the environment. The onsite RPM must be ready to immediately advise plant management on each of the aforementioned items with little or no offsite interaction. The onsite RPM would also have a "feel" for difficult operational/maintenance programs of which an offsite person would have only secondhand knowledge. Finally, Regulatory Guide 8.8 has recently been reviewed by industry at which time no comment was made with respect to the "onsite" RPM meeting the qualifications of Regulatory Guide 1.8. Accordingly, your request regarding the location of the RPM is denied. This position has been discussed with and understood by members of your staff.

OFFICE ▶						
SURNAME ▶						
DATE ▶						

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original signed by

William P. Gammill, Acting Assistant
Director for Operating Reactor Projects
Division of Operating Reactors

Enclosures:
1. Amendment No. 40 to DPR-35
2. Safety Evaluation
3. Notice

cc w/enclosures:
see next page

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cc:

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

BOSTON EDISON COMPANY

DOCKET NO. 50-293

PILGRIM NUCLEAR POWER STATION UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 40
License No. DPR-35

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The portions of the applications for amendment by Boston Edison Company (the licensee) dated July 6 and September 27, 1979, for which an amendment is hereby granted, comply 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 applications, 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.

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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 License No. DPR-35 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 40, 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

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 4, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 40

FACILITY OPERATING LICENSE NO. DPR-35

DOCKET NO. 50-293

Revise Appendix A as follows:

Remove	Insert
32	32
49	49
87	87
137c	137c
137d	137d
137e	137e
160	160
208	208
208a	-
210	210
212	212

TABLE 4.1.2
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration Test (5)	Minimum Frequency (2)
IRM High Flux	C	Comparison to APRM on Controlled Shutdowns	Note (4)
		Full Calibration	once/operating cycle
APRM High Flux			
Output Signal	B	Heat Balance	Once every 3 Days
Flow Bias Signal	B	Internal Power and Flow Test	Each Refueling Outage
LPRM Signal	B	TIP System Traverse	Every 1000 Effective Full Power Hours
High Reactor Pressure	A	Standard Pressure Source	Every 3 Months
High Drywell Pressure	A	Standard Pressure Source	Every 3 Months
Reactor Low Water Level	A	Pressure Standard	Every 3 Months
High Water Level in Scram Discharge Volume	A	Note (6)	Note (6)
Turbine Condenser Low Vacuum	A	Standard Vacuum Source	Every 3 Months
Main Steam Line Isolation Valve Closure	A	Note (6)	Note (6)
Main Steam Line High Radiation	B	Standard Current Source (3)	Every 3 Months
Turbine First Stage Pressure Permissive	A	Standard Pressure Source	Every 6 Months
Turbine Control Valve Fast Closure	A	Standard Pressure Source	Every 3 Months
Turbine Stop Valve Closure	A	Note (6)	Note (6)
Reactor Pressure Permissive	A	Standard Pressure Source	Every 6 Months

PNPS

TABLE 3.2.B (Cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum # of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Remarks
1	Core Spray Pump Start Timer	$0 < t < 1$ sec.	In conjunction with loss of power initiates sequential starting of CSCS pumps
1	LPCI Pump Start Timer	$4 < t < 6$ sec.	
1	LPCI Pump Start Timer	$9 < t < 11$ sec.	
1	Auto Blowdown Timer	≥ 90, ≤ 120 sec.	In conjunction with Low Low Reactor Water Level, High Drywell Pressure and LPCI or Core Spray Pump running interlock, initiates Auto Blowdown.
2	RHR (LPCI) Pump Discharge Pressure Interlock	150 ± 10 psig	Defers ADS actuation pending confirmation of Low Pressure core cooling system operation. (LPCI or Core Spray Pump running interlock.
2	Core Spray Pump Discharge Pressure Interlock	150 ± 10 psig	
2	Emergency Bus Voltage Relay	20-25% of rated voltage resets at less than 50%	1. Permits closure of the Diesel Generator to an unloaded emergency bus. 2. Permits starting of CSCS 4 kV motors.

3.3 and 4.3

BASES:

A. Reactivity Limitation

1. The core reactivity limitation is a restriction to be applied principally to the design of new fuel which may be loaded in the core or into a particular refueling pattern. Satisfaction of the limitation can only be demonstrated at the time of loading and must be such that it will apply to the entire subsequent fuel cycle. The generalized form is that the reactivity of the core loading will be limited so the core can be made subcritical by at least R + 0.25% Δk at the time of the test, with the strongest control rod fully withdrawn and all others fully inserted. The value of R in %Δk is the amount by which the core reactivity, at any time in the operating cycle, is calculated to be greater than at the time of the check; i.e., the initial loading. R must be a positive quantity or zero. A core which contains temporary control or other burnable neutron absorbers may have a reactivity characteristic which increases with core lifetime, goes through a maximum and then decreases thereafter.

The value of R is the difference between the calculated core reactivity at the beginning of the operating cycle and the calculated value of core reactivity any time later in the cycle where it would be greater than at the beginning. The value of R shall include the potential shutdown margin loss assuming full B_4C settling in all inverted poison tubes present in the core. A new value of R must be determined for each full cycle.

The 0.25%Δk in the expression R + 0.25%Δk is provided as a finite, demonstrable, subcriticality margin. This margin is demonstrated by full withdrawal of the strongest rod and partial withdrawal of an adjacent rod to a position calculated to insert at least R + 0.25%Δk in reactivity, or by an insequence, xenon-free cold critical measurement to demonstrate at least R + 0.25% Δk in reactivity with the most reactive control rod fully withdrawn. Observation of subcriticality in this condition assures subcriticality with not only the strongest rod fully withdrawn but at least an R + 0.25% Δk margin beyond this.

Table 3.6.1

SAFETY RELATED SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	Location	Elevation	Snubber in High Radiation Area During Shutdown	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
SS-1-10-1	Main Steam Line	42'			X (Drywell)	
SS-1-10-2	Main Steam Line	42'			X (Drywell)	
SS-1-10-3	Main Steam Line	42'			X (Drywell)	
SS-1-10-4	Main Steam Line	42'			X (Drywell)	
SS-1-10-5	Main Steam Line	42'			X (Drywell)	
SS-1-10-6	Main Steam Line	42'			X (Drywell)	
SS-1-10-7	Main Steam Line	42'			X (Drywell)	
SS-1-10-8	Main Steam Line	42'			X (Drywell)	
SS-1-10-9	Main Steam Line	42'			X (Drywell)	
SS-1-10-10	Main Steam Line	42'			X (Drywell)	
SS-1-10-11	Main Steam Line	42'			X (Drywell)	
SS-1-10-12	Main Steam Line	42'			X (Drywell)	
SS-6-10-6	Feedwater Sys.	41'			X (Drywell)	
SS-6-10-7	Feedwater Sys.	41'			X (Drywell)	
SS-6-10-8	Feedwater Sys.	44'			X (Drywell)	
SS-6-10-9	Feedwater Sys.	41'			X (Drywell)	
SS-6-10-10	Feedwater Sys.	44'			X (Drywell)	
SS-10-30-1	RHR System	52'			X (Drywell)	
SS-10-20-2	RHR System	52'			X (Drywell)	
SS-10-20-3	RHR System	52'			X (Drywell)	
SS-10-20-4	RHR System	52'			X (Drywell)	
SS-10-30-5	RHR System	24'			X (Drywell)	
SS-10-30-6	RHR System	24'			X (Drywell)	
SS-10-20-7	RHR System	24'			X (Drywell)	
SS-10-20-8	RHR System	24'			X (Drywell)	
SS-10-3-9	RHR System	87'			X (Drywell)	
SS-10-3-10	RHR System	90'			X (Drywell)	
SS-2-20-1	Recir. System	42'	X		X (Drywell)	
SS-2-20-2	Recir. System	42'	X		X (Drywell)	
SS-2-20-3	Recir. System	42'	X		X (Drywell)	
SS-2-20-4	Recir. System	42'	X		X (Drywell)	
SS-2-30-5	Recir. System	15'	X		X (Drywell)	
SS-2-30-6	Recir. System	15'	X		X (Drywell)	
SS-2-30-7	Recir. System	15'	X		X (Drywell)	

Table 3.6.1

SAFETY RELATED SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	Location	Elevation	Snubber in High Radiation Area During Shutdown	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
SS-2-30-8	Recir. System	15'	X		X (Drywell)	
SS-2-30-9	Recir. System	11'	X		X (Drywell)	
SS-2-30-10	Recir. System	11'	X		X (Drywell)	
SS-2-30-11	Recir. System	27'	X		X (Drywell)	
SS-2-30-12	Recir. System	27'	X		X (Drywell)	
SS-2-30-13	Recir. System	27'	X		X (Drywell)	
SS-2-30-14	Recir. System	27'	X		X (Drywell)	
SS-2-30-15	Recir. System	27'	X		X (Drywell)	
SS-2-30-16	Recir. System	27'	X		X (Drywell)	
SS-2-20-19	Recir. System	16'	X		X (Drywell)	
SS-2-20-20	Recir. System	16'	X		X (Drywell)	
SS-2-20-21	Recir. System	19'	X		X (Drywell)	
SS-2-20-22	Recir. System	16'	X		X (Drywell)	
SS-2-50-23	Recir. System	17'	X		X (Drywell)	
SS-2-20-24	Recir. System	18'	X		X (Drywell)	
SS-2-20-25	Recir. System	16'	X		X (Drywell)	
SS-2-50-26	Recir. System	16'	X		X (Drywell)	

Table 3.6.1

SAFETY RELATED SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	Location	Elevation	Snubber in High Radiation Area During Shutdown	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
SS-6-10-1	Feedwater System	42'			X (Drywell)	
SS-6-10-2	Feedwater System	42'			X (Drywell)	
SS-6-10-3	Feedwater System	42'			X (Drywell)	
SS-6-10-4	Feedwater System	42'			X (Drywell)	
SS-6-10-5	Feedwater System	42'			X (Drywell)	
SS-13-3-1	RCIC	38'			X (Drywell)	
SS-13-3-2	RCIC	38'			X (Drywell)	
SS-14-3-1	Core Spray	65'			X (Drywell)	
SS-14-3-2	Core Spray	65'			X (Drywell)	
SS-14-3-3	Core Spray	65'			X (Drywell)	
SS-14-3-4	Core Spray	65'			X (Drywell)	
SS-23-10-1	H.P.C.I.	42'			X (Drywell)	
SS-23-10-2	H.P.C.I.	42'			X (Drywell)	
S-23-3-30	H.P.C.I.	-3'09"				X H.P.C.I. Quadrant
S-23-3-31	H.P.C.I.	-3'09"				X H.P.C.I. Quadrant
S-23-10-32	H.P.C.I.	-3'09"				X H.P.C.I. Quadrant
S-23-3-33	H.P.C.I.	-3'09"				X H.P.C.I. Quadrant
S-23-10-34	H.P.C.I.	-6'				X H.P.C.I. Quadrant
S-23-10-35	H.P.C.I.	-6'				X H.P.C.I. Quadrant
S-23-3-36	H.P.C.I.	-3'09"				X H.P.C.I. Quadrant
S-23-3-37	H.P.C.I.	-3'09"				X H.P.C.I. Quadrant
S-10-3-43	RHR	-3'06"				X RHR Pump Room
S-10-20-44	RHR	-3'06"				X RHR Pump Room
S-30-3-45	RBCCW	83'5"				X Reactor Building
S-10-10-46	RHR	6"				X Torus Compartment

Modifications to this Table due to changes in high radiation areas should be submitted to the NRC as part of the next license amendment.

3.7.D Primary Containment Isolation Valves

1. During reactor power operating conditions, all isolation valves listed in Table 3.7.1 and all instrument line flow check valves shall be operable except as specified in 3.7.D.2.

2. In the event any isolation valve specified in Table 3.7.1 becomes inoperable, reactor power operation may continue provided at least one valve in each line having an inoperable valve is in the mode corresponding to the isolated condition.

3. If Specification 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours.

4.7.D Primary Containment Isolation Valves

1. The primary containment isolation valves surveillance shall be performed as follows:

   a. At least once per operating cycle the operable isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation and closure times.

   b. At least once per quarter:

      (1) All normally open power operated isolation valves (except for the main steam line power-operated isolation valves) shall be fully closed and reopened.

      (2) Trip main steam isolation valves individually and verify closure time.

   c. At least twice per week the main steam-line power-operated isolation valves shall be exercised by partial closure and subsequent reopening.

   d. At least once per operating cycle the operability of the reactor coolant system instrument line flow check valves shall be verified.

2. Whenever an isolation valve listed in table 3.7.1 is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

The Pilgrim Station Manager shall be responsible for overall facility operation. In his absence the Assistant Station Manager or an alternate designated in writing by the Station Manager shall assume this responsibility.

6.2 ORGANIZATION

A. OFFSITE

The Company organization for station management and technical support shall be as shown on Figure 6.2.1.

B. FACILITY

The Facility organization shall be as shown on Figure 6.2.2 and:

1. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2.1.

2. At least one licensed Operator shall be in the control room when fuel is in the reactor.

3. At least two licensed Operators shall be present in the control room during reactor startup, scheduled reactor shutdown and during recovery from reactor trips.

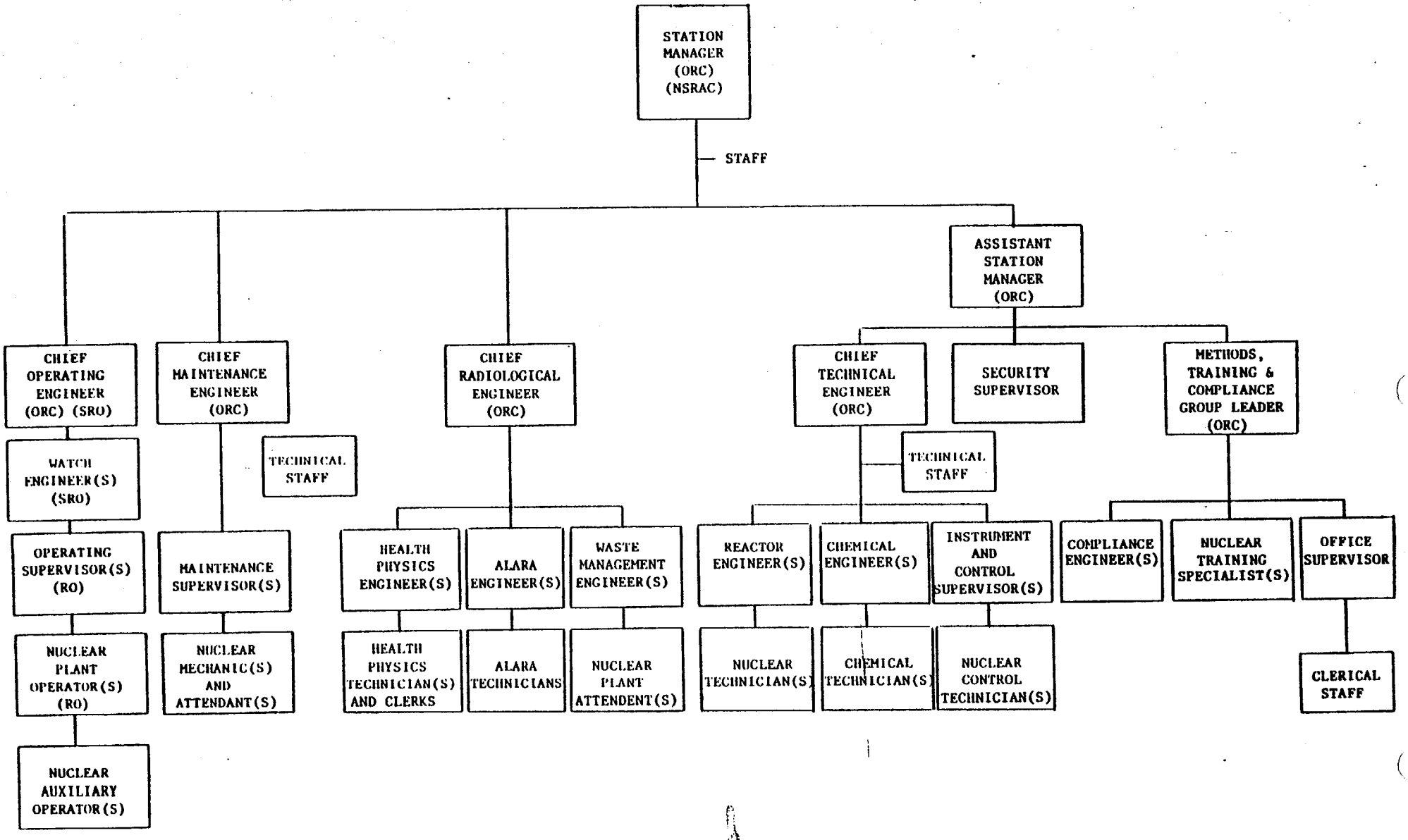
4. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

5. ALL CORE ALTERATIONS performed while fuel is in the reactor vessel after the initial fuel loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

6. The organization of the administration of the Pilgrim I Fire Prevention and Protection Program shall be as shown on Figure 6.2.3. A Fire Brigade of 5 members including the Fire Chief shall be maintained on site at all times. This excludes 3 members of the minimum shift crew necessary for safe shutdown and any personnel required for other essential functions during a fire emergency.

6.3 FACILITY STAFF QUALIFICATIONS

The qualifications with regard to educational and experience backgrounds of the facility staff at the time of appointment to the active position shall meet the requirements as described in the American National Standards Institute N18.1-1971, "Selection and Training of Personnel for Nuclear Power Plants." In addition, the individual performing the function of Radiation Protection Manager shall meet or exceed the qualifications of Regulatory Guide 1.8, September, 1975.



CODE: NSRAC - MEMBER OF NSRAC
ORC - MEMBER OF ORC
RO - NRC REACTOR OPERATOR LICENSE
SRO - NRC SENIOR REACTOR OPERATOR LICENSE

PILGRIM 1 STATION ORGANIZATION
FIGURE 6.2.2

6.5 REVIEW AND AUDIT

A. OPERATIONS REVIEW COMMITTEE (ORC)

1. FUNCTION

The ORC shall function to advise the Pilgrim Station Manager on all matters related to nuclear safety.

2. COMPOSITION

The ORC shall be composed of the:

Chairman/	Station Manager
Vice Chairman	Assistant Station Manager
Member:	Methods, Compliance & Training Group Leader
Member:	Chief Operating Engineer
Member:	Chief Technical Engineer
Member:	Chief Maintenance Engineer
Member:	Chief Radiological Engineer

3. ALTERNATES

Alternate members shall be appointed in writing by the ORC Chairman to serve on a temporary basis; however, no more than two alternates shall participate in an ORC quorum at any one time.

4. MEETING FREQUENCY

The ORC shall meet at least once per calendar month and as convened by the ORC Chairman.

5. QUORUM

A quorum of the ORC shall consist of the Chairman/Vice Chairman and two members including alternates.

6. RESPONSIBILITIES

The ORC shall be responsible for:

a. Review of 1) all procedures required by Specification 6.8 and changes thereto, 2) any other proposed procedures or changes thereto that affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to the Technical Specifications.

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

e. Investigation of all violations of the Technical Specifications and shall prepare and forward a report covering evaluation and recommendations to prevent recurrence to the Nuclear Operations Manager and to the NSRAC Chairman.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 40 TO FACILITY OPERATING LICENSE NO. DPR-35

BOSTON EDISON COMPANY

PILGRIM NUCLEAR POWER STATION UNIT NO. 1

DOCKET NO. 50-293

Introduction

By application dated July 6, 1979, Boston Edison Company (BECo or licensee) proposed an amendment to the Technical Specifications in several areas, each of which was described in an attachment to the letter. Attachment A concerning a change to the method of calibrating the APRM system, and Attachment B concerning the duration of IPCLRT were not addressed in this SER for reasons explained in the amendment issuing letter.

On September 27, 1979, BECo proposed an Administrative TS change which supplemented Attachment D to the July 6, 1979 submittal. Both submittals are evaluated in this report.

I. Update Safety Related Snubbers

1. Discussion

Attachment C proposed four changes in Table 3.6.1. They are:

1.1. The deletion of snubbers SS-2-10-17, SS-2-10-18, and SS-3-3-1 from Table 3.6.1.

1.2. The change of designation of snubbers SS-6-10-1 to SS-6-10-10, and SS-2-20-5 to SS-2-30-5.

1.3. The change of elevation to 42' for snubbers SS-20-20-1, SS-2-20-2, SS-2-20-3, and SS-2-20-4.

1.4. The change of prefix SS to S for snubbers located outside the drywell.

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2. Evaluation

2.1 Deletion of Snubbers

The proposed deletion of three snubbers from Table 3.6.1 is an updating effort to reflect plant revisions made in 1976 and 1978. Snubbers SS-2-10-17 and SS-2-10-18 were attached to the reactor recirculation pump discharge valve 4" by-pass lines. Both lines and snubbers were removed in 1976 as reported in the 1976 annual 10 CFR 50.59 report. Snubber SS-3-3-1 was attached to the control rod drive system return line. This line and snubber were removed as reported in the 1978 annual 10 CFR 50.59 report. The updating effort on Table 3.6.1 has no impact on plant safety.

2.2 Change of Designation of Snubbers

The re-number of snubber designation, corrects some previous typographical errors. It has no impact on plant safety.

2.3 Change of Elevation for Snubbers

The change of elevation of snubbers reflects the actual position. It has no impact on plant safety.

2.4 Change of Prefix for Snubbers

The change of prefix for snubber designation offers better identification of the snubber locations. It has no impact on plant safety.

3. Conclusion

Since the proposed revisions add to the clarity of Table 3.6.1, and do not compromise plant safety, their implementation should be permitted.

II. Station Organization Chart and ORC Composition

1. Discussion

Attachment D proposed restructuring Figure 6.2.2, "Pilgrim 1 Station Organization" to accommodate organization changes at the Station. Specifically, several new positions were created to increase the station management's attention toward the Health Physics Program. In particular, the position of Chief Radiological Engineer was created to head the Health Physics

Program. In addition, the Composition of the Operations Review Committee (ORC) was revised to include the Chief Radiological Engineer as a member. The September 27, 1979 submittal proposed further restructuring of Figure 6.2.2 to provide for an Assistant Station Manager who will also serve as Vice Chairman of the ORC.

2. Evaluation

We agree that the creation of new positions including Senior ALARA Engineer, ALARA Engineer, ALARA Health Physics Technicians, and Senior Waste Management Engineer under the Chief Radiological Engineer should improve the Health Physics Program at Pilgrim. Further, we concur with the change in the organization to improve the workload distribution through the Station, including the position of Assistant Station Manager and additions to ORC membership.

3. Conclusion

We conclude that the proposed changes to the Pilgrim 1 Station Organization are designed to strengthen the station management and are therefore, acceptable.

III. Response Time Requirement for ADS

1. Discussion

Attachment E proposed a change in the setting of the Automatic Depressurization System (ADS) blowdown timer from "120 ± 5 seconds" to "greater than or equal to 90 to less than or equal to 120 seconds". The purpose of this change is to assure that the actual value used is less than that used in the ECCS analysis (120 seconds) and to provide margin in setpoint to account for calibration errors and drift.

2. Evaluation

The lower value is sufficiently greater than the High Pressure Coolant Injection System (HPCI) starting time (25 seconds) and (in the licensee's opinion) the response time of a trained operator to maintain the probability of false initiation of the ADS at an acceptably lower level. The proposed ADS timer trip setpoint is consistent with the Standard Technical Specifications (STS) for General Electric Boiling Water Reactors, NUREG 0123 Revision 2 August 1979 and satisfies the requirements of IEEE Std. 279-1971.

3. Conclusion

We find that this change will not compromise the effectiveness of the ADS to provide its design protection and that it is desireable to assure compatibility between the Technical Specification and LOCA analysis. We therefore conclude that this change is acceptable.

IV. Reactivity Shutdown Margin Demonstration

1. Discussion

Attachment F requested a change in the wording of the Technical Specification bases to allow shutdown margin demonstration by a method different than that listed in the bases for Specification 3.3.A.1. The two rod method (pulling the strongest worth rod and a diagonally adjacent rod to a specified position) is listed in the bases as the method of demonstrating shutdown margin at PNPS.

2. Evaluation

The licensee desires to have the option of using the dispersed uniform insequence control rod withdrawal sequence method of demonstrating shutdown reactivity margin. The reason is to avoid highly peaked flux distributions which can lead to sudden unexpected criticals with fast periods. The dispersed uniform withdrawal sequence is designed specifically to minimize rod worths, thus substantially reducing the probability of high reactivity insertion incidents. The licensee proposed to perform this test at beginning of life (BOL) fuel cycle conditions, cold xenon-free. This is consistent with the NUREG-0123 Revision 2 August 1979 bases.

3. Conclusion

We agree that the proposed reactivity shutdown margin demonstration offers a more conservative approach and is acceptable.

V. IRM High Flux Instrument Calibration

1. Discussion

In Attachment G, the licensee has proposed a Full Calibration of the IRM High Flux Channels once per operating cycle. This was done to clarify the functional and calibration test requirements of the Nuclear Instrumentation Intermediate Range Monitoring (IRM) system, in response to an IE Inspection Report 50-293/78-18 and followup correspondence dated September 21, 1978.

2. Evaluation

We have reviewed the licensee's proposal in conjunction with the current TS requirements to verify overall conformance with GE STS. We find that the proposed change is consistent with STS requirements for IRM Channel Calibration and is therefore acceptable. Further, we find that the current TS requirement for IRM Funcational Test in Table 4.1.1 is consistent with STS functional test requirements. Also, the current TS requirement for IRM Calibration Test in Table 4.1.2 is the equivalent of the STS Channel Check, except that the minimum frequency is not specified. The STS requires an IRM Channel Check (Instrument Check, as defined in PNPS TS) at least once every 12 hours. We verified that station procedures require the equivalent of an IRM Channel Check each shift. This is acceptable until such time as BECo adopts the STS for Pilgrim Nuclear Power Station.

3. Conclusion

We conclude that the licensee's proposal is consistent with GE STS and satisfies the requirements of IEEE Std. 279-1971 and is therefore, acceptable.

VI. MSIV Test Requirements

1. Discussion

Attachment H requested removal of the current power restriction requirement of 50% reactor power when verifying closure time of Main Steam Isolation Valves. During a previous refueling outage at Pilgrim, a pressure averaging manifold was installed to allow the testing of MSIV's and the turbine stop and control valves at full power levels and thus eliminate the need for costly power reductions. Amendment No. 34 to Facility Operating License No. DPR-35 dated September 19, 1978 changed the trip level setting for the High Flow Main Steam Line instruments from $\leq$120% to $\leq$140% of rated steam flow. The setpoint was changed to decrease the possibility of an automatic closure of all MSIV's and the attendant reactor scram which can occur when performing valve closure time testing.

2. Evaluation

The licensee has stated that the proposed change was intended to be included in Amendment No. 34 and was inadvertently omitted. We have reviewed the previous Safety Evaluation associated with Amendment No. 34 and agree that it satisfies all concerns related to safety.

3. Conclusion

We conclude that the removal of the power restriction for testing MSIV's is acceptable.

Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR Section 51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendment.

Conclusion

We have concluded based on the considerations discussed above that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: February 4, 1980

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-293

BOSTON EDISON COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 40 to Facility Operating License No. DPR-35, issued to Boston Edison Company (the licensee), which revised the Technical Specifications for operation of the Pilgrim Nuclear Power Station Unit No. 1 (the facility) located near Plymouth, Massachusetts. The amendment is effective as of its date of issuance.

This amendment changes the Technical Specifications to (1) update Table 3.6.1 for safety related snubbers, (2) revise the station organization chart and the Operations Review Committee (ORC) composition, (3) reduce the response time requirement for ADS, (4) provide an alternate method of demonstrating reactivity margin, (5) add calibration and test frequencies to the IRM High Flux Instrument Channels, and (6) remove the power restriction requirement for testing MSIV closure times.

The application for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR Section 51.5(d)(4), an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of the amendment.

For further details with respect to this action, see (1) the applications for amendment dated July 6 and September 27, 1979, (2) Amendment No. 40 to License No. DPR-35, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Plymouth Public Library on North Street in Plymouth, Massachusetts 02360. A single copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 4 day of February 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors