FEBRUARY 22 1980

Docket No. 50-293

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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. 41 to Operating License No. DPR-35 for the Pilgrim Nuclear Power Station. This amendment consists of changes to the Technical Specifications in response to your request dated November 21, 1979 as supplemented by letter dated December 7, 1979. This amendment changes the Technical Specifications to make provisions for Multiple Control Rod Removal during Refueling Operations.

During our review of the proposed Technical Specifications, we determined that certain changes to your request were necessary to conform with NRC requirements. These changes were discussed with and agreed to by members of your staff, who were, by the way, exceptionally cooperative and responsive in resolving the questions we raised.

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

Sincerely,
Original Signed by
T. A. Ippolito

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

Enclosures:
1. Amendment No. 41 to DPR-35
2. Safety Evaluation
3. Notice

cc w/encls:
See next page

concur as to form of FR Notice & Amend only

CP

OFFICE	DOR:ORB#3	DOR:ORB#3	DOR:ORP	OELD	DOR:ORB#3	
SURNAME	SSheppard	JHannon:ms	WPGammill	Olmstead	TAIppolito	
DATE	2/15/80	2/19/80	2/19/80	2/20/80	2/21/80	

cc:

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Energy Facilities Siting Council
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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. 41
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 November 21, 1977 as supplemented December 7, 1979, 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 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. 41 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

8003110347

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 22, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 41

FACILITY OPERATING LICENSE NO. DPR-35

DOCKET NO. 50-293

Revise Appendix A as follows:

Remove	Insert
203	203
-	203a
205	205
	208a (correct an admin. error in Amendment 40)

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
2. The SRM shall have a minimum of 3 cps except as specified in 3 and 4 below. 3. Prior to spiral unloading, the SRM's shall have an initial count rate of ≥ 3 cps. During spiral unloading, the count rate on the SRM's may drop below 3 cps. 4. During spiral reload, each control cell shall have at least one assembly with a minimum exposure of 1000 MWD/t.	**Spiral Reload** During spiral reload, SRM operability will be verified by using a portable external source every 12 hours until the required amount of fuel is loaded to maintain 3 cps. As an alternative to the above, up to two fuel assemblies will be loaded in different cells containing control blades around each SRM to obtain the required 3 cps. Until these assemblies have loaded, the cps requirement is not necessary.
C. **Spent Fuel Pool Water Level** Whenever irradiated fuel is stored in the spent fuel pool, the pool water level shall be maintained at or above 33 feet.	C. **Spent Fuel Pool Water Level** Whenever irradiated fuel is stored in the spent fuel pool, the water level shall be recorded daily.
D. **Multiple Control Rod Removal** Any number of control rods and/or control rod drive mechanisms may be removed from the reactor pressure vessel provided that at least the following requirements are satisfied until all control rods and control rod drive mechanisms are reinstalled and all control rods are fully inserted in the core.	D. **Multiple Control Rod Removal** Within 4 hours prior to the start of removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and at least once per 24 hours thereafter until all control rods and control rod drive mechanisms are reinstalled and all control rods are fully inserted in the core, verify that:
a. The reactor mode switch is operable and locked in the Refuel position per Specification 3.10.A, except that the Refuel position "one rod out" interlock may be bypassed, as required, for those control rods and/or control rod drive mechanisms to be removed, after the fuel assemblies have been removed as specified below	a. The reactor mode switch is operable and locked in the Refuel position per Specification 3.10.A.
b. The source range monitors (SRM) are operable per Specification 3.3.B.4.	b. The SRM channels are operable per Specification 3.3.B.4.
c. The Reactivity Margin requirements of Specification 3.3.A.1 are satisfied.	c. The Reactivity Margin requirements of Specification 3.3.A.1 are satisfied.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
d. All control rods in a 3x3 array centered on each of the control rods being removed are fully inserted and electrically or hydraulically disarmed, or have the surrounding four fuel assemblies removed from the core cell.	d. All control rods in 3 x 3 array centered on each of the control rods removed or being removed are fully inserted and electrically or hydraulically disarmed, or have the surrounding four fuel assemblies removed.
e. All other control rods are fully inserted.	e. All other control rods are fully inserted.
f. The four fuel assemblies are removed from the core cell surrounding each control rod or control rod drive mechanism to be removed from the core and/or reactor vessel.	f. The four fuel assemblies surrounding each control rod and/or control rod drive mechanism that is to be removed from the reactor vessel at the same time are removed from the core and/or reactor vessel.

3.10 BASES

B. Core Monitoring

The SRM's are provided to monitor the core during periods of station shutdown and to guide the operator during refueling operations and station startup. Requiring two operable SRM's in or adjacent to any core quadrant where fuel or control rods are being moved assures adequate monitoring of that quadrant during such alterations. The requirement of 3 counts per second provides assurance that neutron flux is being monitored and insures that startup is conducted only if the source range flux level is above the minimum assumed in the control rod drop accident.

The limiting conditions for operation of the SRM subsystem of the Neutron Monitoring System are derived from the Station Nuclear Safety Operational Analysis (Appendix G) and a functional analysis of the neutron monitoring system. The specification is based on the Operational Nuclear Safety Requirements in subsection 7.5.10 of the Safety Analysis Report.

C. Spent Fuel Pool Water Level

To assure that there is adequate water to shield and cool the irradiated fuel assemblies stored in the pool, a minimum pool water level is established. The minimum water level of 33 feet is established because it would be a significant change from the normal level (-1 foot) and is well above the level to assure adequate cooling.

D. Multiple Control Rod Removal

These specifications ensure that maintenance or repair of control rods or rod drives will be persormed under conditions that limit the probability of inadvertent criticality. The requirement that the fuel assemblies in the cell controlled by the control rod be removed from the reactor core before the interlock can be bypassed insures that withdrawal of another control rod does not result in inadvertent criticality. Each control rod essentially provides reactivity control for the fuel assemblies in the cell associated with the control rod. Thus, removel of an entire cell (fuel assemblies plus control rod) results in a lower reactivity potential of the core.

4.10 BASES

A. Refueling Interlocks

Complete functional testing of all refueling interlocks before any refueling outage will provide positive indication that the interlocks operate in the situations for which they were designed. By loading each hoist with a weight equal to the fuel assembly, positioning the refueling platform, and withdrawing control rods, the interlocks can be subjected to valid operational tests. Where redundancy is provided in the logic circuitry, tests can be performed to assure that each redundant logic element can independently perform its functions.

B. Core Monitoring

Requiring the SRM's to be functionally tested prior to any core alteration assures that the SRM's will be operable at the start of that alteration. The daily response check of the SRM's ensures their continued operability.

6.4 <u>TRAINING</u>

A. A retraining and replacement training program for the facility staff shall be maintained under the direction of the Pilgrim Station Manager.

B. A retraining program for the licensed operators shall be maintained under the direction of the Senior Nuclear Training Specialist and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix A of 10 CFR Part 55.

C. A training program for the Fire Brigade shall be maintained under the direction of the Fire Protection and Prevention Officer and shall meet or exceed the requirements of Section 27 of the NFPA Code 1975. Training sessions will be held quarterly.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 41 TO FACILITY OPERATING LICENSE NO. DPR-35

BOSTON EDISON COMPANY

PILGRIM NUCLEAR POWER STATION UNIT NO. 1

DOCKET NO. 50-293

1. Introduction

By letter dated November 21, 1979,[1] Boston Edison Company (the licensee) has requested an amendment to the Technical Specifications for the Pilgrim Nuclear Power Station. The effect of the amendment would be to allow multiple control blade removal with the reactor in the refuel mode. Analyses supporting this amendment were submitted on December 7, 1979.[2]

2. Discussion

2.1 Motivation

When the reactor mode switch is in the "refuel" position, the refueling interlocks will allow one control blade, but no more than one, to be withdrawn. (When fuel assemblies are being removed or inserted, additional interlocks on the fuel handling equipment will block withdrawal of even one blade.)

The licensee wishes to override this interlock. This will allow scheduled maintenance to be done on several control rod drives simultaneously. In addition to shortened downtime, with a probable economic benefit to the licensee, we note that personnel exposure (man-rem) should be reduced by this change.

2.2 Safety Concern

The safety concern is to ensure that the reactor remains subcritical, i.e., that shutdown margin is preserved. The licensee proposes to do this by removing the four fuel assemblies surrounding each blade to be withdrawn, before overriding the interlock on that particular blade. Although it would seem obvious that removing the fuel surrounding a blade would introduce negative reactivity, it is not necessarily true that fuel removal would more than compensate for removal of the control rod. Therefore, the licensee has calculated the change in shutdown margin for a wide range of configurations and fuel enrichment to demonstrate that, for Pilgrim 1, fuel removal does indeed more than compensate for control blade removal.

3. Evaluation

3.1 Codes Used

The basic methodology used in the analysis of the multiple control rod removal configurations relies on the use of CASMO, a two-dimensional lattice code which calculates macroscopic cross sections for a variety of fuel assembly conditions, and PDQ-7, a fine mesh diffusion theory code used here in a two-dimensional mode to calculate K_{eff} for the various core configurations.

To provide confidence in this methodology, two sets of benchmark calculations were performed. First, a set of CASMO calculations for a broad group of uniform lattices was checked against experiment. Since the output of CASMO forms the basis input to the other calculations, a check of this nature is essential.

Second, the 2D PDQ code was used to calculate the K_{eff} of actual BOC cold critical configurations in the Pilgrim reactor. Because of the two-dimensional nature of these PDQ calculations, the code can only consider fully inserted and fully withdrawn blades. Thus, the K_{eff} of the actual core configurations was based on two PDQ calculations, one with all partially withdrawn rods assumed full out and one with all partially withdrawn rods assumed full in, and the assumption was made that the reactivity worth of the partially withdrawn rods was linearly related to the number of notches of withdrawal.

These benchmark calculations are very limited. The CASMO benchmark calculations were performed only on uniform lattices, and thus were not completely realistic, since a BWR core contains water gaps, channel boxes and control blades. In addition, this assumption of linear incremental rod worth in the PDQ benchmark calculations is very approximate, since actual incremental rod worths are quite non-linear. This error can be considered conservative in the sense that the ultimate use of these codes does not involve partially withdrawn blades, and thus any error estimate derived from the benchmark calculations should overestimate the error in the end use calculations. Because of this, and because of the limited number of benchmark cases, we cannot at this time extend general approval of this code package, even though we consider it adequate for this "trend" analysis, as described below.

3.2 Analysis

The goal of this analysis was to show that the withdrawal of one or more control cells from the Pilgrim 1 core will lead to a more subcritical state. The licensee calculated K_{eff} for the BOC-5 core, then re-calculated

K_{eff} assuming one or more empty control cells. Fifteen cases were run, with the number of empty cells varying from one to 16. The cases studied included a full range of positions of the empty cells within the core and positions of the empty cells with respect to one another. All the calculations assumed a cold xenon-free core, a conservative assumption. In every case, the core became more subcritical.

Because of the gadolinia loading in a BWR core, the most reactive state occurs after BOC. To conservatively bound this effect, the licensee recalculated two of the cases, this time with the gadolinia removed from the fresh fuel. We agree that this should bound the reactivity swing actually experienced during the cycle. Again, the core becomes more subcritical in each case.

The licensee then attempted to bound all possible loadings by calculating a hypothetical core containing assemblies of the maximum enrichment currently available, with the gadolinia removed. This results in a far more reactive core than could be loaded in actual practice. Six cases were studied, and in each case the removal of blade plus fuel made the core reactivity decrease.

Finally, the licensee studied this same highly reactive core (in quarter-core geometry) with one cell containing four highly burned assemblies. Thus, core reactivity is maximized, the rod worth is maximized, and the negative reactivity effect of fuel removal is minimized. Here too the core became (slightly) more subcritical.

We agree that the cases studied should bound any configurations encountered in actual practice. Although we cannot give general approval to the codes used at this time, we do agree that the safety concern of interest here (preservation of shutdown margin) has been adequately addressed because:

- Shutdown margin increased in <u>all</u> cases studied. It is very unlikely that any random error in the calculation of K_{eff} could have led to this result.

- Because all cases involve the differencing of two calculations, most systematic errors should cancel out. The major exception is the cross sections of the water-filled cavity, for which the licensee has provided a separate sensitivity study.

- A completely separate calculation, done by the NRC staff, has shown the same trend.(3)

- Another completely separate calculation, on another docket, also supports this trend.(4)

Therefore, we find these calculations acceptable.

3.3 Technical Specification Implementation

Shutdown margin, as defined in the plant Technical Specifications and explicitly addressed in Specifications 3.3.A.1 and 4.3.A.1, is required to be maintained with the highest worth control blade stuck out of the core. The calculations described above did not consider the effect of withdrawing a control blade in addition to the blade associated with the water-filled cavity, nor did the licensee's proposed change address this situation. Therefore, we will require that the 8 blades surrounding any such cavity be fully inserted and disarmed. Blades which are inserted and disarmed need not be considered in the search for the strongest rod, consistent with Specifications 4.3.A.1 and 3.3.A.2.b. Thus, with the consent of the licensee, the proposed specifications(1) have been altered as follows:

Replace Proposed Specifications 3.10.D.d and 3.10.D.e with:

d. All control rods in a 3x3 array centered on each of the control rods being removed are fully inserted and electrically or hydraulically disarmed, or have the surrounding four fuel assemblies removed from the core cell.

e. All other control rods are fully inserted.

f. The four fuel assemblies are removed from the core cell surrounding each control rod or control rod drive mechanism to be removed from the core and/or reactor vessel.

Replace Proposed Specifications 4.10.D.d and 4.10.D.e by:

d. All control rods in 3x3 array centered on each of the control rods removed or being removed are fully inserted and electrically or hydraulically disarmed, or have the surrounding four fuel assemblies removed.

e. All other control rods are fully inserted.

f. The four fuel assemblies surrounding each control rod and/or control rod drive mechanism that is to be removed from the reactor vessel at the same time are removed from the core and/or reactor vessel.

4. Summary

We have examined the calculational methods used and found them acceptable for this purpose. We have examined the set of cases studied and found it adequate. Therefore, with the change in the amendment detailed in Section 3.3 above, we find this amendment to be acceptable.

5. 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.

6. 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.

7. References

1. Letter, J. E. Howard (Boston Edison) to T. A. Ippolito (NRC), dated November 21, 1979.

2. Letter, G. C. Andognini (Boston Edison) to T. A. Ippolito (NRC), dated December 7, 1979.

3. Memo, L. Kopp (Core Performance Branch/DSS/NRC) to D. Fieno (Core Performance Branch/DSS/NRC), dated July 18, 1979.

4. Technical Specification Change Request No. 51 to Provisional Operating License No. DPR-16, Oyster Creek Nuclear Generating Station, Docket Number 50-219, dated November 19, 1976.

Dated: February 22, 1980

7590-01

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. 41 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 make provisions for Multiple Control Rod Removal during Refueling Operations.

The application for the amendment complies 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 application for amendment dated November 21, 1979 as supplemented December 7, 1979, (2) Amendment No. 41 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, NW., Washington, D. C., and at the Plymouth Public Library on North Street in Plymouth, Massachuseets 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 22nd day of February 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

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