OCT 10 1984

Docket No. 50-293

Mr. William D. Harrington
Senior Vice President, Nuclear
Boston Edison Company
800 Boylston Street
Boston, Massachusetts 02199

Dear Mr. Harrington:

The Commission has issued the enclosed Amendment No. 82 to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station. The amendment changes the Technical Specifications in response to your request dated July 30, 1984.

These changes to the Technical Specifications add the schedule for withdrawal of material surveillance capsules from the reactor pressure vessel (RPV). They also revise the RPV thermal and pressurization limit curves to reflect the RT_{NDT} shift determined from actual testing of the first capsule removed. The adjusted curves included in this amendment are for 6.68 and 8.0 effective full power years of reactor operation. Subsequent periods of operation will be addressed in a future licensing action. The related limiting condition for operation (LCO) has also been modified to require the operator to take certain actions if the thermal and pressurization limits are not met.

A copy of our Safety Evaluation relative to your request is also enclosed.

Sincerely,

Paul H. Leech, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 82 to License No. DPR-35
2. Safety Evaluation

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See next page

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Mr. William D. Harrington
Boston Edison Company
Pilgrim Nuclear Power Station

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

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 82
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 July 30, 1984 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-35 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 82 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the
Technical Specifications

Date of Issuance: October 10, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 82

FACILITY OPERATING LICENSE NO. DPR-35

DOCKET NO. 50-293

Remove	Insert
123	123
124	124
---	124A
128	128
128A	128A
139	139
---	139A

3.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the operating status of the reactor coolant system.

Objective:

To assure the integrity and safe operation of the reactor coolant system.

Specification:

A. Thermal and Pressurization Limitations

1. The average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100°F/hr when averaged over a one-hour period except when the vessel temperatures are above 450°F. The shell flange to shell temperature differential shall not exceed 145°F.

2. The reactor vessel shall not be pressurized for hydrostatic and/or leakage tests, and critical core operation shall not be conducted unless the reactor vessel temperature is above that defined by the appropriate curves on Figures 3.6.1 and 3.6.2. In the event this requirement is not met, achieve stable reactor conditions with reactor vessel temperature above that defined by the appropriate curve and obtain an engineering evaluation to determine the appropriate course of action to take.

4.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the periodic examination and testing requirements for the reactor cooling system.

Objective:

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification:

A. Thermal and Pressurization Limitations

1. During heatups and cooldowns, the following temperatures shall be permanently logged at least every 15 minutes until the difference between any two readings taken over a 45 minute period is less than 5°F.

   a. Reactor vessel shell adjacent to shell flange

   b. Reactor vessel shell flange

   c. Recirculation loops A and B

2. Reactor vessel shell temperature and reactor coolant pressure shall be permanently logged at least every 15 minutes whenever the shell temperature is below 220°F and the reactor vessel is not vented.

   Test specimens of the reactor vessel base, weld and heat affected zone metal subjected to the highest fluence of greater than 1 Mev neutrons shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The specimens and sample program shall conform to the requirements of ASTM E 185-66. Selected

LIMITING CONDITION FOR OPERATION

3.6.A <u>Thermal and Pressurization Limitations</u> (Cont'd)

3. The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel head flange and the head is greater than 55°F.

4. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other.

5. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and the bottom head drain are within 145°F.

6. <u>Thermal-Hydraulic Stability</u>

   Core thermal power shall not exceed 25% of rated thermal power without forced recirculation.

B. <u>Coolant Chemistry</u>

1. The reactor coolant system radioactivity concentration in water shall not exceed 20 microcuries of total iodine per ml of water.

2. The reactor coolant water shall not exceed the following limits with steaming rates less than 100,000 pounds per hour, except as specified in 3.6.B.3:

   Conductivity ... 2 μmho/cm

   Chloride ion ... 0.1 ppm

SURVEILLANCE REQUIREMENTS

4.6.A <u>Thermal and Pressurization Limitations</u> (Cont'd)

neutron flux specimens shall be removed at the frequency required by Table 4.6.3 and tested to experimentally verify adjustments to Figures 3.6.1 and 3.6.2 for predicted NDTT irradiation shifts.

3. When the reactor vessel head bolting studs are tensioned and the reactor is in a Cold Condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

4. Prior to and during startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged.

5. Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged.

B. <u>Coolant Chemistry</u>

1. a. A reactor coolant sample shall be taken at least every 96 hours and analyzed for radioactivity content.

   b. Isotopic analysis of a reactor coolant sample shall be made at least once per month.

2. During startups and at steaming rates less than 100,000 pounds per hour, a sample of reactor coolant shall be taken every four hours and analyzed for chlorine content.

TABLE 4.6.3
REACTOR VESSEL MATERIAL
SURVEILLANCE PROGRAM WITHDRAWAL SCHEDULE

Capsule Number	Effective Full Power Years (EFPY)	Fluence (n/cm^2) (1/4 T)	RT_{NDT} (weld metal) (°F)
1	4.17	1.8×10^{17}	55
2	15 (approx.)	6.3×10^{17} (approx.)	91
3	End of Life	1.4×10^{18} (approx.)	136

FIGURE 3.6.1
PILGRIM REACTOR VESSEL
PRESSURE - TEMPERATURE LIMITS
HYDROSTATIC AND LEAK TESTS

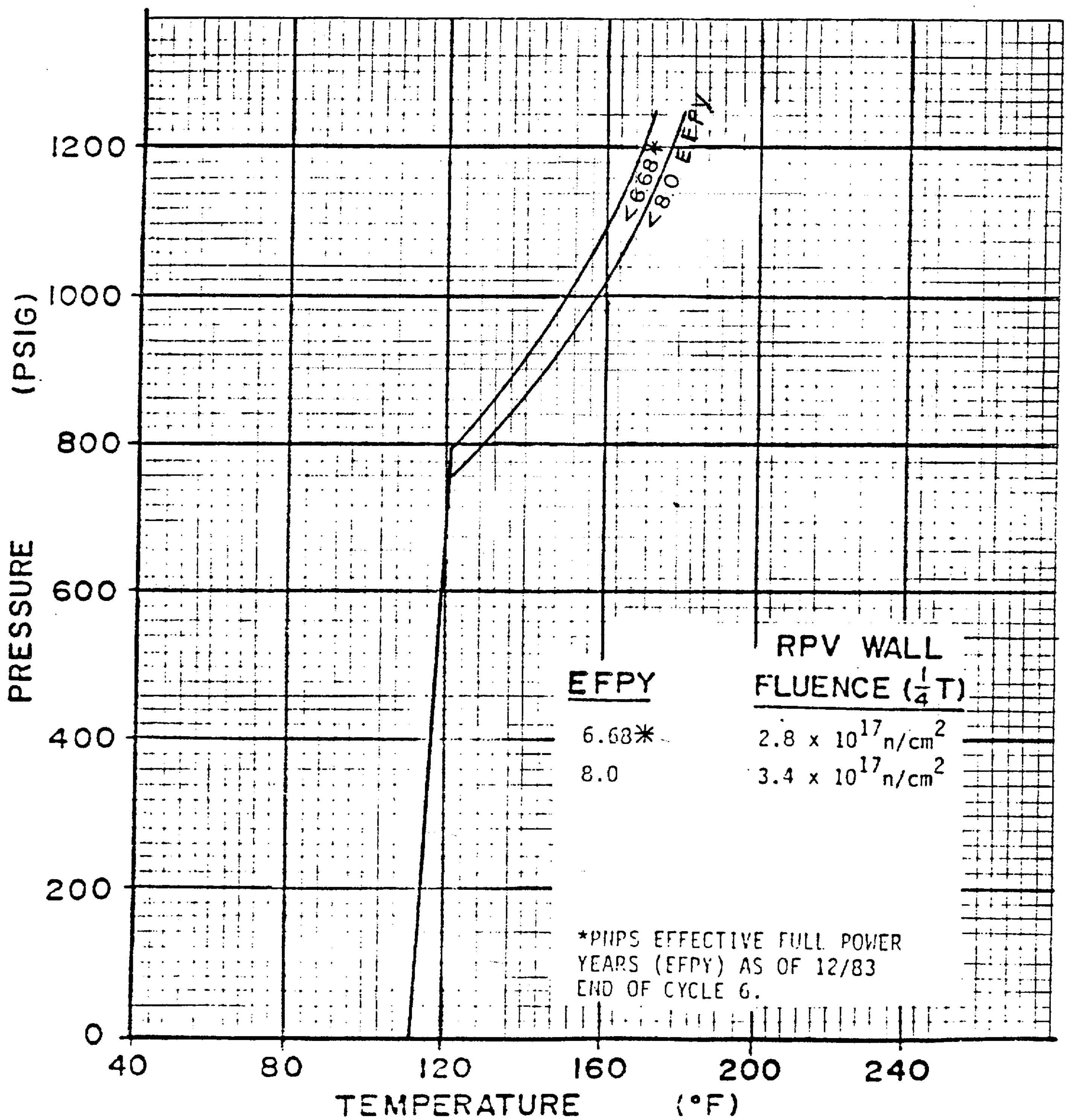
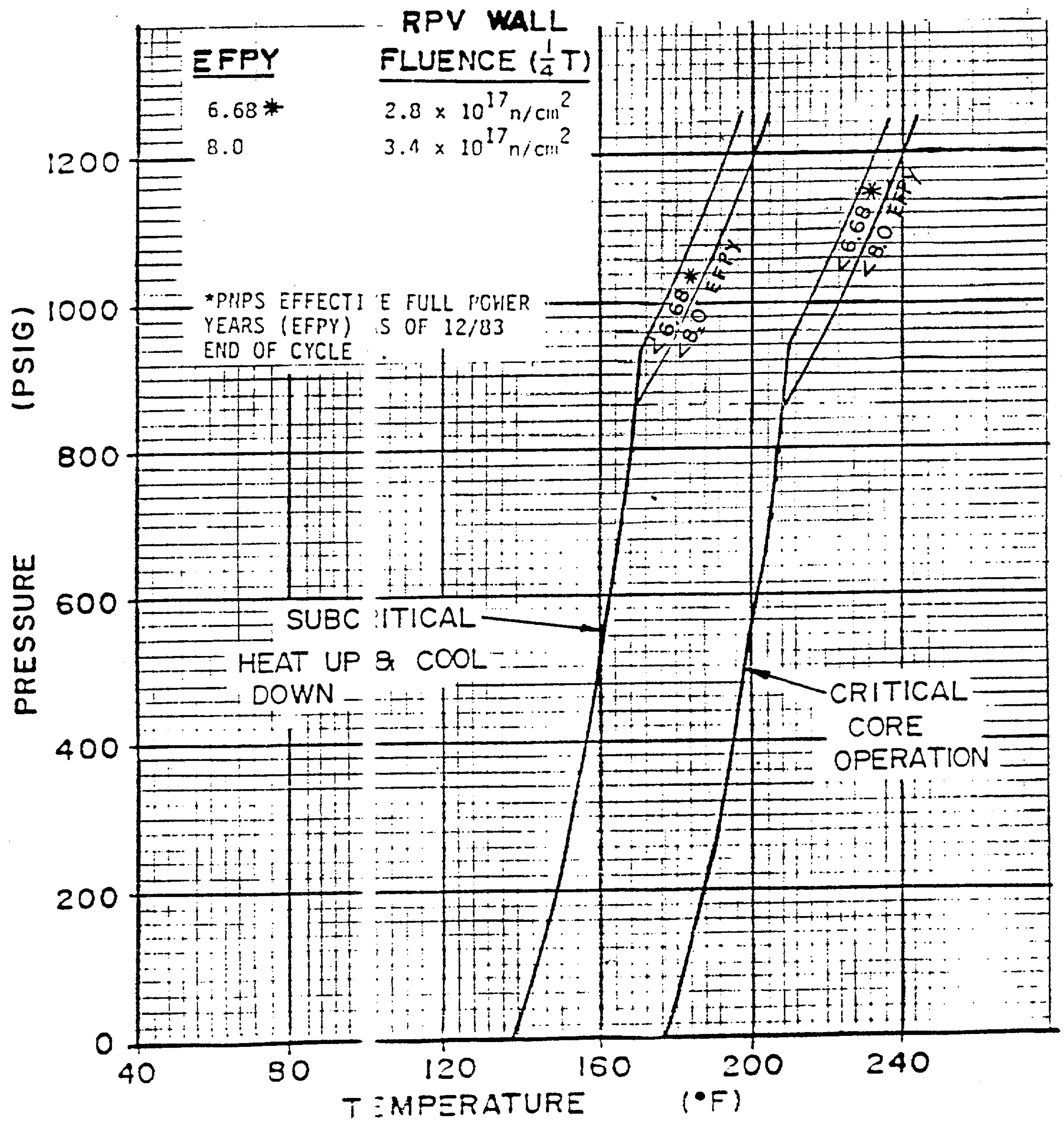


FIGURE 3.6.2

PILGRIM REACTOR VESSEL
PRESSURE - TEMPERATURE LIMITS
SUBCRITICAL/CRITICAL HEAT UP & COOL DOWN



Bases:

3.6.A and 4.6.A

Thermal and Pressurization Limitations (Cont'd)

The reactor coolant system is a primary barrier against the release of fission products to the environs. In order to provide assurance that this barrier is maintained at a high degree of integrity, restrictions have been placed on the operating conditions to which it can be subjected.

Appendix G to 10CFR50 defines the temperature-pressurization restrictions for hydrostatic and leak tests, pressurization, and critical operation. These limits have been calculated for Pilgrim and are contained in Figures 3.6.1 and 3.6.2.

For Pilgrim pressure-temperature restrictions, two locations in the reactor vessel are limiting. The closure region controls at lower pressures and the beltline controls at higher pressures.

The nil-ductility transition (NDT) temperature is defined as the temperature below which ferritic steel breaks in a brittle rather than ductile manner. Radiation exposure from fast neutrons (>1 mev) above about 10^{17} nvt may shift the NDT temperature of the vessel metal above the initial value. Impact tests from the first material surveillance capsule removed from the reactor vessel have established the magnitude of the RT_{NDT} shift for the beltline. The shift, which is greatest for the weld metal, is tabulated below for various fluence levels and EFPY of operation:

EFPY	RPV Wall Fluence (1/4T)	RT_{NDT}
6.68*	2.8×10^{17} n/cm^2	61°F
8.0	3.4×10^{17} n/cm^2	68°F

*PNPS Effective Full Power Years (EFPY) as of end of Cycle 6 (12/83)

Neutron flux wires and samples of vessel material are installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The wires and samples will be periodically removed and tested to experimentally verify the values used for Figures 3.6.1 and 3.6.2. The withdrawal schedule of Table 4.6.3 has been established as required by 10CFR50, Appendix H.

The pressure-temperature limitations of Figures 3.6.1 and 3.6.2 applicable to the beltline reflect an initial RT_{NDT} of 0°F. This initial value is based

Bases:

3.6.A and 4.6.A

Thermal and Pressurization Limitations (Cont'd)

on unirradiated test data adjusted for specimen orientation in accordance with USNRC Branch Technical Position MTEB 5-2.

The pressure-temperature limitations of Figures 3.6.1 and 3.6.2 applicable to the closure region reflect an RT_{NDT} of -5°F, also based on test data adjusted for specimen orientation. The curves apply to 100% bolt preload condition, but are conservative for lesser bolt preload conditions.

For critical core operation when the water level is within the normal range for power operation and the pressure is less than 20% of the preservice system hydrostatic test pressure (313 psi), the minimum permissible temperature of the highly stressed regions of the closure flange is RT_{NDT} + 60 = 55°F.

The closure region is more limiting than the feedwater nozzle with regards to both stress intensity and RT_{NDT}. Therefore the pressure-temperature limits of the closure are controlling.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 82 TO FACILITY OPERATING LICENSE NO. DPR-35

BOSTON EDISON COMPANY

PILGRIM NUCLEAR POWER STATION

DOCKET NO. 50-293

1.0 Introduction

In a letter from W. D. Harrington to D. B. Vassallo dated July 30, 1984, the Boston Edison Company (BECo) requested changes to the pressure temperature limits and surveillance capsule withdrawal schedule for the Pilgrim Nuclear Power Station (PNPS). BECo has provided five sets of pressure-temperature limit curves (Leak and Hydrostatic Test, Heat-up and Cooldown, Critical Core Operation) which they indicate meet the safety margins of Appendix G, 10 CFR 50, for a period of time corresponding to 6.68 effective full power years (EFPY), 8.0 EFPY, 10.0 EFPY, 12.0 EFPY and 14.3 EFPY, respectively. BECo indicates that the revised withdrawal schedule meets the requirements of Appendix H, 10 CFR 50.

2.0 Evaluation

Pressure-temperature limits must be calculated in accordance with the requirements of Appendix G, 10 CFR 50, which became effective on July 26, 1983. Pressure-temperature limits that are calculated in accordance with the requirements of Appendix G are dependent upon the initial RT_{NDT} for the limiting materials in the beltline and closure flange regions of the reactor vessel and the increase in RT_{NDT} resulting from neutron irradiation damage to the limiting beltline material.

The PNPS reactor vessel was procured prior to the issuance of the Appendix G. However, the PNPS reactor vessel materials must meet the safety margins and testing requirements of the regulation. Appendix G requires that samples from each reactor vessel material be fracture toughness tested to determine their initial (unirradiated) RT_{NDT}.

In order to determine the RT_{NDT} of a material, both drop weight and Charpy V-notch (CVN) tests are required. For base metal, the CVN test specimen must be oriented perpendicular to the principal working direction of the plate. The materials in the PNPS reactor vessel were tested to an ASME Code edition and addenda that did not require sufficient testing to determine each material's RT_{NDT}. However, the licensee has used the available test results and the method of estimating RT_{NDT} recommended in Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements," to determine the initial RT_{NDT} for weld and base metal in the closure flange and beltline regions of the PNPS reactor vessel. The licensee's analysis is documented in Teledyne Engineering Services Technical Report TR-6052-1, Rev. 2, July 19, 1984, which was submitted with the requested change in the Technical Specifications. The unirradiated RT_{NDT} for the beltline welds and limiting plate were estimated as 0°F and -3°F respectively. The RT_{NDT} for the limiting plate in the closure flange region was estimated as -5°F.

The Pilgrim surveillance material data is documented in the Southwest Research Institute Report SWRI Project No. 02-5951, "Pilgrim Nuclear Power Station Unit 1 Reactor Vessel Irradiation Surveillance Program," dated July 1981. The amount of copper in the plate and weld surveillance material was reported as .14 percent and .16 percent, respectively. The increase in RT_{NDT} resulting from neutron irradiation damage was estimated by the licensee by extrapolating the test data points from the Pilgrim surveillance program at the rate of embrittlement reported in Regulatory Guide (R.G.) 1.99, Rev. 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," dated April 1977. This is the method recommended by R.G. 1.99, Rev. 1, when the surveillance material is judged most likely to be controlling with regard to radiation damage.

The pressure temperature limit curves proposed by the licensee are more conservative than those in the existing technical specification. Hence, the proposed curves provide more margin against brittle fracture of the reactor vessel than the curves presently in the Pilgrim technical specification. This additional margin permits us to conclude that the Pilgrim reactor vessel may be operated using the proposed curves marked 8 EFPY until the next refueling outage; i.e., shutdown following the end of cycle. Additional closure flange and beltline region materials data must be provided by the licensee prior to the next refueling outage to permit us to complete our evaluation of the remaining curves proposed by the licensee.

Appendix H of 10 CFR 50 requires that the surveillance capsule withdrawal schedule comply with ASTM E-185-82. The withdrawal schedule proposed by the licensee for the remaining two capsules meets the withdrawal schedule requirements of ASTM E-185-82. Hence, it satisfies Appendix H and may be incorporated into the Pilgrim Technical Specification.

3.0 Environmental Consideration

This amendment involves changes in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding.

Accordingly, this 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 this amendment.

4.0 Conclusion

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: B. Elliott

Dated: October 10, 1984