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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 140 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 11, 1991, the Boston Edison Company (the licensee) submitted a request for changes to the Pilgrim Nuclear Power Station, Technical Specifications (TS). The requested changes would revise the pressure/temperature (P/T) limits in the Pilgrim Technical Specifications, Section 3.6. The proposed P/T limits were requested for 10, 11, 13, 15, 20, and 32 effective full power years (EFPY). In October 1991, the EFPY is about 9.1. The proposed P/T limits were developed using Regulatory Guide (RG) 1.99, Revision 2. Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations," recommends RG 1.99, Rev. 2, be used in calculating P/T limits, unless the use of different methods can be justified. A P/T limits for the bottom head of the reactor vessel were also requested.

To evaluate the P/T limits, the staff uses the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Rev. 2; Standard Review Plan (SRP) Section 5.3.2; and Generic Letter 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide Technical Specifications for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the Technical Specifications. The P/T limits are among the limiting conditions of operation in the Technical Specifications for all commercial nuclear plants in the U.S. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). ' neric Letter 88-11 requested that licensees and permittees use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline. By letters dated August 5 and October 4, 1991, the licensee requested Bases changes to Sections 3.1, 3.2, 3.3 and 4.3, 3.5.C, D & E, 3.9 and 3.10.

2.0 EVALUATION

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the Pilgrim reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff has determined that the material with the highest ART at 32 EFPY was the lower intermediate shell axial weld (1-338A, B, and C) with 0.13% copper (Cu), 1.06% nickel (Ni), and an initial $RT_{ndt}$ of -35°F.

The licensee has removed one surveillance capsule from Pilgrim. The results from capsule 1 were published in Southwest Research Institute Report SwRI 02-5951. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material, weld 1-338A, B, and C, the staff calculated the ART to be 90.9°F at 1/4T (T = reactor vessel beltline thickness) and 70.2°F for 3/4T at 32 EFPY. The staff used a neutron fluence of 9.8E17 n/cm$^2$ at 1/4T and 4.9E17 n/cmY2H at 3/4T. The ART was determined by Section 1 of RG 1.99, Rev. 2, because only one capsule was removed from the Pilgrim reactor pressure vessel.

The licensee used the method in RG 1.99, Rev. 2, to calculate an ART of 91°F at 32 EFPY at 1/4T for the same limiting weld metal. The staff judges that the licensee's ART of 91°F is more conservative than the staff's ART of 90.9°F, and it is acceptable. Substituting the ART of 91°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for 32 EFPY for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50. The staff also verified that P/T limits for 10, 11, 13, 15, and 20 EFPYs meet the Appendix G requirements.

In addition to beltline mate als, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the pre-service system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Paragraph IV.A.3 of Appendix G states "an exception may be made for boiling water reactor vessels when water level is within the normal range for power operation and the pressure is less than 20 percent of the pre-service system hydrostatic test pressure. In this case the minimum permissible temperature is 60°F (33°C) above the reference temperature of the closure flange regions that are highly stressed by the bolt preload." Based on the flange reference temperature of -5°F, the staff has determined that the proposed P/T limits for the 11, 13, 15, 20, and 32 EFPYs satisfy Section IV.A.2 of Appendix G.

In regard to the proposed bottom head P/T limits, the staff believes that because the bottom head of the reactor vessel does not receive significant amount of neutron fluence, embrittlement due to irradiation is not of major concern. The reference temperature calculations for the reactor beltline materials as prescribed in RG 1.99, Rev. 2 are not applicable to the bottom head P/T limits. The licensee calculated the stresses of the bottom head materials due to internal pressure, startup and cooldown transients, deadweight, and seismic loadings. The maximum stress locations are located at the junction between the lower torus and the support skirt and at control rod penetrations. From the maximum stresses, stress intensity factors and P/T limits were calculated based on ASME Code, Section III, Appendix G and 10 CFR 50, Appendix G. The staff finds that the licensee's calculation satisfy 10 CFR 50, Appendix G. However, to safeguard the structural integrity of the reactor beltline materials, the licensee must ensure that the pressure and temperature readings from the P/T sensors at the reactor vessel beltline region must be within the acceptable region of the beltline P/T limit curves when the bottom head P/T limits are being used during heatup and cooldown. The bottom head P/T limits must follow the same heatup and cooldown rate, 100 degrees F per hour, as that of the beltline P/T limits.

## 3.0 STATE CONSULTATION

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

## 4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (56 FR 31429). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

## 5.0 CONCLUSION

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

## 6.0 REFERENCES

1. Regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Materials, Revision 2, May 1988

2. NUREG-0800, Standard Review Plan Section 5.3.2: Pressure-Temperature Limits

3. Letter from G. W. Davis (BECo) to USNRC Document Control Desk, Subject: Proposed Changes to the Reactor Pressure Vessel Thermal and Pressurization Technical Specification Limits, June 11, 1991

4. E. B. Norris, "Pilgrim Nuclear Power Station Unit 1 Reactor Vessel Irradiation Surveillance Program, SHRI 02-5951," July 1981

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Date: January 29, 1992