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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION SUPPORTING

AMENDMENT NO. 153 TO FACILITY OPERATING LICENSE NO. DPR-44

PHILADELPHIA ELECTRIC COMPANY

PUBLIC SERVICE ELECTRIC AND GAS COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 2

DOCKET NO. 50-277

1.0 INTRODUCTION

By letter dated May 15, 1989, Philadelphia Electric Company requested an amendment to Facility Operating License No. DPR-44 for the Peach Bottom Atomic Power Station, Unit No. 2. The proposed amendment modifies the pressure-temperature limits for the reactor vessel.

In response to Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations," the Philadelphia Electric Company (the licensee) requested permission to revise the pressure/temperature (P/T) limits in the Peach Bottom Atomic Power Station, Unit 2 (hereinafter, Peach Bottom 2) Technical Specifications, Section 3/4.6. The purpose of the revision is to change the effectiveness of the P/T limits for 32 effective full power years (EFPY). The proposed P/T limits were based on Regulatory Guide (RG) 1.99, Revision 2. The proposed revision provides up-to-date P/T limits for the operation of the reactor coolant system during heatup, cool-down, criticality, and hydrotest.

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, Revision 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). Generic Letter 88-11 requested that licensees and permittees use the methods in RG 1.99, Revision 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.

2.0 EVALUATION

The staff evaluated the effect of neutron irradiation embrittlement on each belt-line material in the Peach Bottom 2 reactor vessel. The amount of neutron irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The material with the highest ART at 32 EFPY was the lower-intermediate shell plate C2873-1 with 0.12% copper (Cu) and 0.57% nickel (Ni), and an initial RT of -6°F.

The licensee has removed one surveillance capsule from Peach Bottom 2. The results from that surveillance capsule were published in General Electric Report SASR 88-24, DRF B13-01445, which is an attachment to a letter from J. W. Gallagher to T. E. Murley dated May 13, 1988. The surveillance capsule contained Charpy impact specimens and tensile specimens which were made from base metal, weld metal, and HAZ metal.

For the limiting beltline material, C2873-1, the staff calculated the ART to be 51°F at 1/4T (T = reactor vessel beltline thickness) for 32 EFPY. The staff calculated the ART by the method described in Section 1 of RG 1.99, Rev. 2 because only one surveillance capsule had been withdrawn from the Peach Bottom 2 reactor vessel. The licensee calculated the same 51°F for the ART. Substituting the ART of 51°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

In addition to beltline materials, 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.2 of Appendix G states that when the pressure exceeds 20% of the preservice 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. Figure 3.6.2 of the proposed Technical Specifications for Peach Bottom 2 shows that the temperature for heatup or cooldown following nuclear shutdown is approximately 165°F at 300 psig. Figure 3.6.1 of the proposed Technical Specifications for Peach Bottom 2 shows that the minimum temperature for pressure tests required by Section XI of the ASME Code is 100°F at 312 psig. Based on the flange reference temperature of 10°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G for normal operation, hydrostatic pressure and leak tests.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life (EOL) be above 50 ft-lb. The initial USE for the limiting beltline material, the lower-intermediate shell plate metal (C2873-1), was not supplied. However, the calculated USE for the surveillance base metal (C2761-2) at EOL is 84.5 ft-lb, which is higher than the Appendix G EOL USE requirement. The surveillance base metal (C2761-2) was produced by the same manufacturer to the same ASTM specification as the limiting beltline material, and has copper and nickel contents that are very close to those of the limiting beltline material (0.11% Cu and 0.54% Ni for C2761-2 vs 0.12% Cu and 0.57% Ni for C2873-1). Based on this comparison, the staff believes that the EOL USE of the limiting beltline material (C2873-1) meets the Appendix G 50 ft-lb requirement.

The staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 32 EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The licensee's submittal also satisfies Generic Letter 88-11 because the licensee used the method in RG 1.99, Rev. 2, to calculate the ART. Hence, the proposed P/T limits may be incorporated into the Peach Bottom 2 Technical Specifications.

The licensee also proposed certain administrative changes to the Technical Specification pages involved with the changes discussed above. These changes include the deletion of Figure 3.6.4 which provides information on estimated transition temperature shift relative to fluence; rewording T.S. 3.6.A.3 to more accurately describe the vessel materials and appurtenances involved; revision of the "neutron flux specimen" terminology in T.S. 4.6.A.2 to "surveillance specimen"; revisions to T.S. page 144 to reflect removal and testing of a surveillance capsule and deletion of Figure 3.6.4; related changes in the List of Figures; and minor format and typographical changes on page 143 and 144. Related changes to the T.S. Bases are also proposed.

The staff finds that these proposed changes reflect the results of material analyses conducted as part of the reactor coolant pressure boundary material surveillance program. These changes are consistent with the proposed changes to the reactor vessel pressure-temperature limits and are thus acceptable. The staff also finds the proposed addition of Figure 3.6.5 to the List of Figures properly reflects its addition in a previously approved license amendment, and is thus acceptable.

3.0 ENVIRONMENTAL CONSIDERATIONS

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to the 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 nor environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (54 FR 31116) on July 26, 1989 and consulted with the Commonwealth of Pennsylvania. No public comments were received and the Commonwealth of Pennsylvania did not have any comments.

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

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Dated: October 25, 1989

TABLE 1

The NRC Staff Calculated Adjusted Reference Temperature for the Limiting Reactor Beltline Material at Peach Bottom Atomic Power Station, Unit 2.

Limiting Beltline Material:		Lower-intermediate shell plate
Code No.:		C2873-1
Copper Content:		0.12%
Nickel Content:		0.57%
Initial Reference Temperature:		-6°F
Reactor Vessel Beltline Thickness (in.)		6.31
Reactor Vessel Beltline Inside Radius (in.)		125.5
Chemistry Factor (CF) Used in Calculation		82.4
Neutron Fluence n/cm^2 at 32 EFPY:		
	At I.D.	1.0E18
	At 1/4T	0.69E18
	AT 3/4T	0.33E18
Fluence Factor		
	At I.D.	0.417
	At 1/4T	0.347
	At 3/4T	0.231
Margin		28.5°F
ART at 1/4T at 32 EFPY:		51°F (Licensee calculated 51°F)
ART at 3/4T at 32 EFPY		32°F (Licensee did not provide an ART for 3/4T in the GE report, SA 88-24)