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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 36 TO FACILITY OPERATING LICENSE NO. NPF-73

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

THE TOLEDO EDISON COMPANY

BEAVER VALLEY POWER STATION, UNIT NO. 2

DOCKET NO. 50-412

1.0 INTRODUCTION

By letter dated June 11, 1990, (Ref. 1) Duquesne Light Company proposed a revision to the pressure/temperature (P-T) limits in the Beaver Valley Power Station, Unit 2 (BVPS-2) Technical Specifications, Section 3.4 (Ref. 2). This revision changes the P-T limits from 5 to 10 effective full power years (EFPY). The proposed P-T limits were developed based on Regulatory Guide (RG) 1.99, Revision 2, (Ref. 3) and they provide limits for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest.

To evaluate the P-T limits, we used 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 (Ref. 4); and Generic Letter 88-11 (Ref. 5).

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, 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 withdraw periodically surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which 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 DISCUSSION AND EVALUATION

We have evaluated the effect of neutron irradiation embrittlement on each beltline material in the Beaver Valley Unit 2 reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. We have determined that the material with the highest ART at 10 EFPY at the 1/4T and 3/4T (T = reactor vessel beltline thickness) locations was intermediate shell plate B9004-1 with 0.07% copper (Cu), 0.53% nickel (Ni), and an initial RT_{ndt} of 60°F.

Duquesne Light Company has removed surveillance capsule U from the BVPS-2 reactor vessel after 1.24 EFPY. The encapsulated specimens were tested, and the results were reported in WCAP-12406. We have reviewed WCAP-12406 and conclude that it satisfies the reporting requirement in 10 CFR 50 Appendix H. We also have determined that all surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material, plate B9004-1, we calculated the ART to be 140.4°F at 1/4T and 128.6°F at 3/4T. The staff used a neutron fluence of 1.22E19 n/cm^2 at 1/4T and 4.66E18 n/cm^2 at 3/4T.

Duquesne Light Company used the method in RG 1.99, Rev. 2, to calculate an ART of 140°F at 1/4T and 129°F at 3/4T for the same limiting metal. Substituting the ART of 140.4°F into equations in SRP 5.3.2, we 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 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. Based on the flange reference temperature of 0°F, we have determined that the proposed P-T limits satisfy Section IV.2 of Appendix G.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. The material with the lowest initial USE is plate B9004-2 with an initial USE of 75.5 ft-lb. Using the method in RG 1.99, Rev. 2, the predicted Charpy USE of the plate at the end of life will be greater than 50 ft-lb and, therefore, is acceptable.

We find that the proposed P-T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 10 EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. Duquesne Light Company's submittal also satisfies Generic Letter 88-11 because the method in RG 1.99, Rev. 2, was used to calculate the ART. Hence, the proposed P-T limits as represented on revised Figures 3.4-2 and 3.4-3 may be incorporated into the BVPS-2 Technical Specifications.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment changes 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. We have 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 staff 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, 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.

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 (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.

Dated: January 7, 1991

Principal Contributor: John Tsao

5.0 REFERENCES

1. June 11, 1990, Letter from J. D. Sieber (Duquesne Light Company) to USNRC Document Control Desk, Subject: Beaver Valley Power Station, Unit 2, Proposed Operating License Change Request No. 40 (TAC NO. 76890).

2. Beaver Valley Unit 2 Technical Specifications.

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

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

5. Generic Letter 88-11, NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations, USNRC, July 12, 1988.