Enclosure 1

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

NO. DPR-66

DUQUESNE LIGHT COMPANY

BEAVER VALLEY POWER STATION UNIT 1

DOCKET NO. 50-334

1.0 INTRODUCTION

In response to Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations," Duquesne Light Company (the licensee) proposed to revise the pressure/temperature (P/T) limits in the Beaver Valley Power Station Unit 1 Technical Specifications. The proposal was documented in a letter from the licensee dated November 23, 1988. This proposal also changes the effectiveness of the P/T limits for 9.5 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, cooldown, 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, Rev. 2; Standard Review Plant (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, 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.

2.0 EVALUATION

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the Beaver Valley Unit 1 reactor vessel. The amount of neutron irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff has determined that the material with the highest ART at 9.5 EFPY for Beaver Valley Unit 1 was the lower shell plate (B6607-2) with 0.14% copper (Cu), 0.62% nickel (Ni), and an initial RT_{ndt} of 73°F.

The licensee has removed three surveillance capsules (V, U, and W) from Beaver Valley Unit 1. The results from capsule V were published in Westinghouse Report WCAP-9860; the results from capsule U in WCAP-10867; and the results from capsule W in WCAP-12005. 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 B6607-2, the staff calculated the ART at 9.5 EFPY at 1/4T (T = reactor vessel beltline thickness) to be 201.5°F and 175.6°F at 3/4T. The staff used a fluence of 8.1E18 n/cm^2 for 1/4T and 3.15E18 n/cm^2 for 3/4T. The ART was determined by Section 1 of RG 1.99, Rev. 2.

The licensee used the method in RG 1.99, Rev. 2, to calculate an ART of 202°F at 9.5 EFPY at 1/4T and 176°F at 3/4T for the same limiting plate material. The staff judges that a difference of less than 1°F between the licensee's ART of 202°F and the staff's ART of 201.5°F is acceptable. Substituting the ART of 202°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. Based on the flange reference temperature of 60°F, the staff has 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. Based on data from a surveillance capsule withdrawn at 5.89 EFPY, the measured transverse Charpy USE is 59 ft-lb for the lower shell plate material, B6903-1. This is a 26.2% reduction from the unirradiated value of 80 ft-lb. Using the method in RG 1.99, Rev. 2, the predicted Charpy USE of the lower shell plate material at the end of life will be below 50 ft-lb. The staff will monitor the weld metal Charpy USE from future surveillance capsules. The surveillance capsule data will provide early warning of the decrease in Charpy USE, because the surveillance capsule lead factors are greater than 1.0.

3.0 CONCLUSION

The staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 9.5 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 Beaver Valley Unit 1 Technical Specifications via a future amendment.

4.0 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. November 23, 1988, Letter from J. D. Sieber (DL) to USNRC Document Control Desk, subject: Beaver Valley Power Station, Units 1 and 2 Response to Generic Letter 88-11

4. January 24, 1989, Letter from J. D. Sieber (DL) to USNRC Document Control Desk, subject: Beaver Valley Power Station, Unit 1, Reactor Vessel Capsule W Test Results Report (WCAP-12005)

5. R. S. Boggs et al, "Analysis of Capsule U from the Duquesne Light Company Beaver Valley Unit Reactor Vessel Radiation Surveillance Program, WCAP-10867," Westinghouse Electric Corporation, September 1985

6. October 15, 1981, Letter from J. J. Carey (DL) to S. A. Varga (USNRC), subject: Beaver Valley Power Station, Unit 1, Reactor Vessel Irradiation Specimen Test Report (WCAP-9860)

7. July 21, 1977, Letter from C. N. Dunn (DL) to R. W. Reid (USNRC), subject: Beaver Valley Power Station, Unit 1, Reactor Vessel Material Surveillance Program

8. Beaver Valley Final Safety Analysis Report

5.0 PRINCIPAL CONTRIBUTORS

John Tsao, with contractual assistance from the Idaho National Engineering Laboratory. Report completed in November, 1989.

Enclosure 2

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WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

NO. NPF-73

DUQUESNE LIGHT COMPANY

BEAVER VALLEY POWER STATION, UNIT 2

DOCKET NO. 50-412

1.0 INTRODUCTION

In response to Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations," Duquesne Light Company (the licensee) proposed to revise the pressure/temperature (P/T) limits in the Beaver Valley Power Station, Unit 2 Technical Specifications, Section 3.4. The proposal was documented in a letter from the licensee dated November 23, 1988. This proposal also changes the effectiveness of the P/T limits from 10 to 5 effective full power years (EFPY). The proposed P/T limits were developed based on Section 1 of 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, cooldown, 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, Rev. 2; Standard Review Plant (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, 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.

2.0 EVALUATION

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the Beaver Valley 2 reactor vessel. The amount of neutron irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff has determined that the material with the highest ART at 5 EFPY was the intermediate shell plate with 0.07% copper (Cu), 0.53% nickel (Ni), and an initial RT_{ndt} of 60°F.

The licensee has not withdrawn any surveillance capsules from the Beaver Valley 2 reactor vessel. All six surveillance capsules contain Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material, plate B9004-1, the staff calculated the ART to be 132°F at 1/4T (T = reactor vessel beltline thickness) for 5 EFPY. The neutron fluence at 1/4T was estimated to be 6.3E18 n/cm² for 5 EFPY. The ART was determined by Section 1 of RG 1.99, Rev. 2, because no surveillance capsule data are available.

The licensee used the method in RG 1.99, Rev. 2, to calculate an ART of 132°F at 5 EFPY at 1/4T for the same limiting plate material. Substituting the ART of 132°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. Based on the flange reference temperature of 0°F, the staff has 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 unirradiated USE is intermediate shell plate B9004-2 with 75.5 ft-lb. Based on its 0.07% Cu and Figure 2 of RG 1.99, Rev. 2, the EOL (32 EFPY) USE is predicted to be 53.6 ft-lb. This is greater than 50 ft-lb and, therefore, is acceptable.

3.0 CONCLUSION

The staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 5 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 Beaver Valley Unit 2 Technical Specifications via a future amendment.

4.0 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. November 23, 1988, Letter from J. D. Sieber (DL) to USNRC Document Control Desk, subject: Beaver Valley Power Station, Units 1 and 2 Response to Generic Letter 88-11

4. Beaver Valley Final Safety Analysis Report

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