August 3, 2000

Mr. L. W. Myers
Senior Vice President
Beaver Valley Power Station
Post Office Box 4
Shippingport, PA 15077

SUBJECT: BEAVER VALLEY 2 - ENVIRONMENTAL ASSESSMENT AND FINDING OF NO SIGNIFICANT IMPACT RELATED TO EXEMPTION FROM REQUIREMENTS OF 10 CFR PART 50, SECTION 50.60(a) AND 10 CFR PART 50, APPENDIX G (TAC NO. MA5988)

Dear Mr. Myers:

Enclosed is a copy of the Environmental Assessment and Finding of No Significant Impact related to your request, dated June 17, 1999, for an exemption from the requirements of 10 CFR 50, Section 50.60(a) and 10 CFR Part 50, Appendix G. The proposed exemption would allow application of Code Case N-640 in establishing the reactor vessel pressure limits at low temperatures for Beaver Valley Power Station, Unit No. 2.

The assessment is being forwarded to the Office of the Federal Register for publication.

Sincerely,

/RA/

Daniel S. Collins, Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-412

Enclosure: Environmental Assessment

cc w/encl: See next page

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DATE	7/19/00	7/19/00	7/20/00	7/25/00	8/2/00

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UNITED STATES NUCLEAR REGULATORY COMMISSION

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

ENVIRONMENTAL ASSESSMENT AND FINDING OF

NO SIGNIFICANT IMPACT

The U.S. Nuclear Regulatory Commission (NRC) is considering issuance of an exemption from the requirements of Title 10 of the Code of Federal Regulations (10 CFR) Section 50.60(a), and 10 CFR Part 50, Appendix G, for Facility Operating License No. NPF-73, issued to FirstEnergy Nuclear Operating Company (the licensee), for operation of the Beaver Valley Power Station, Unit No. 2 (BVPS-2), located in Beaver County, Pennsylvania.

ENVIRONMENTAL ASSESSMENT

Identification of the Proposed Action:

Appendix G to 10 CFR Part 50, requires that pressure/temperature (P/T) limits be established for reactor pressure vessels during normal operating and hydrostatic or leak rate testing conditions. Specifically, this regulation states, “The appropriate requirements on both the pressure-temperature limits and the minimum permissible temperature must be met for all conditions.” Additionally, it specifies that the requirements for these limits are contained in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI, Appendix G.

To address provisions of an amendment to the Technical Specification P/T limits, the licensee requested in its submittal dated June 17, 1999, that the NRC staff exempt BVPS-2 from the requirements of 10 CFR Part 50, Section 50.60(a), and 10 CFR Part 50, Appendix G, to allow application of ASME Code Case N-640 in establishing the reactor vessel pressure limits at low temperatures.

Code Case N-640 permits the use of an alternate reference fracture toughness (K_{IC} fracture toughness curve instead of the K_{Ia} fracture toughness curve) for reactor vessel materials in determining the P/T limits. Since the K_{IC} fracture toughness curve shown in ASME, Section XI, Appendix A, Figure A-2200-1 (the K_{IC} fracture toughness curve), provides greater allowable fracture toughness than the corresponding K_{Ia} fracture toughness curve of ASME, Section XI, Appendix G, Figure G-2210-1 (the K_{Ia} fracture toughness curve), using Code Case N-640 for establishing the P/T limits would be less conservative than the methodology currently endorsed by 10 CFR Part 50, Appendix G. Therefore, an exemption is required in order to apply the Code Case. It should be noted that, although Code Case N-640 was incorporated into the ASME Code recently, an exemption is still required because the proposed P/T limits (excluding Code Case N-640) are based on the 1989 edition of the ASME Code.

The proposed action is in accordance with the licensee's application for exemption dated June 17, 1999.

The Need for the Proposed Action:

ASME Code Case N-640 is needed to revise the method used to determine the reactor coolant system (RCS) P/T limits.

The purpose of 10 CFR Part 50, Section 50.60(a), and 10 CFR Part 50, Appendix G, is to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. This is accomplished through these regulations that, in part, specify fracture toughness requirements for ferritic materials of the reactor coolant pressure boundary. Pursuant to 10 CFR Part 50,

Appendix G, it is required that P/T limits for the RCS be at least as conservative as those obtained by applying the methodology of the ASME Code, Section XI, Appendix G.

Current overpressure protection system (OPPS) setpoints produce operational constraints by limiting the P/T range available to the operator to heat up or cool down the plant. The operating window through which the operator heats up and cools down the RCS becomes more restrictive with continued reactor vessel service. Reducing this operating window could potentially have an adverse safety impact by increasing the possibility of inadvertent OPPS actuation due to pressure surges associated with normal plant evolutions such as reactor coolant pump start and swapping operating charging pumps with the RCS in a water-solid condition. The impact on the P/T limits and OPPS setpoints has been evaluated for an increased service period to 15 effective full power years based on ASME Code, Section XI, Appendix G, requirements. The results indicate that OPPS would significantly restrict the ability to perform plant heatup and cooldown, create an unnecessary burden to plant operations, and challenge control of plant evolutions required with OPPS enabled. Continued operation of BVPS-2 with P/T curves developed to satisfy ASME Code, Section XI, Appendix G, requirements without the relief provided by ASME Code Case N-640 would unnecessarily restrict the P/T operating window, especially at low temperature conditions.

Application of ASME Code Case N-640 will provide results which are sufficiently conservative to ensure the integrity of the reactor coolant pressure boundary while providing P/T curves which are not overly restrictive. Implementation of the proposed P/T curves, as allowed by ASME Code Case N-640, does not significantly reduce the margin of safety.

In the associated exemption, the NRC staff has determined that, pursuant to 10 CFR 50.12(a)(2)(ii), the underlying purpose of the regulation will continue to be served by the implementation of ASME Code Case N-640.

Environmental Impacts of the Proposed Action:

The NRC has completed its evaluation of the proposed action and concludes that the proposed action provides adequate margin of safety against brittle failure of the reactor coolant pressure boundary.

The proposed action will not significantly increase the probability or consequences of accidents, no changes are being made in the types of any effluents that may be released off site, and there is no significant increase in occupational or public radiation exposure. Therefore, there are no significant radiological environmental impacts associated with the proposed action.

With regard to potential nonradiological impacts, the proposed action does not involve any historic sites. It does not affect nonradiological plant effluents and has no other environmental impact. Therefore, there are no significant nonradiological environmental impacts associated with the proposed action.

Accordingly, the NRC concludes that there are no significant environmental impacts associated with the proposed action.

Alternatives to the Proposed Action:

As an alternative to the proposed action, the staff considered denial of the proposed action (i.e., the “no-action” alternative). Denial of the application would result in no change in current environmental impacts. The environmental impacts of the proposed action and the alternative action are similar.

Alternative Use of Resources:

This action does not involve the use of any resources not previously considered in the Final Environmental Statement for BVPS-2.

Agencies and Persons Consulted:

In accordance with its stated policy, on July 10, 2000, the staff consulted with the Pennsylvania State official, Mr. L. Ryan of the Pennsylvania Department of Environmental Protection Bureau, Division of Nuclear Safety, regarding the environmental impact of the proposed action. The State official had no comments.

FINDING OF NO SIGNIFICANT IMPACT

On the basis of the environmental assessment, the NRC concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the NRC has determined not to prepare an environmental impact statement for the proposed action.

For further details with respect to the proposed action, see the licensee's letter dated June 17, 1999, which is available for public inspection at the Commission's Public Document Room, The Gelman Building, 2120 L Street, NW., Washington, DC. Publicly available records will be accessible electronically from the ADAMS Public Library component on the NRC Web site, http:\\www.nrc.gov (the Electronic Reading Room).

Dated at Rockville, Maryland, this 3rd day of August, 2000 .

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Daniel S. Collins, Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation