September 6, 2000

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

SUBJECT: BEAVER VALLEY POWER STATION, UNIT 2 - ISSUANCE OF EXEMPTION AND AMENDMENT RE: AMENDED PRESSURE-TEMPERATURE LIMITS (TAC NO. MA5988)

Dear Mr. Myers:

The Commission has issued the enclosed Amendment No. 113 to Facility Operating License No. NPF-73 for the Beaver Valley Power Station, Unit 2 (BVPS-2). This amendment consists of changes to the TSs in response to the application dated June 17, 1999, by Duquesne Light Company, as the then licensee for Beaver Valley Power Station, Unit 2 (BVPS-2), as supplemented by letters dated September 15, 1999, and February 15, and June 29, 2000, which submitted License Amendment Request No. 127.

The amendment approves new low temperature over-pressure protection set points and new heatup and cooldown pressure/temperature (P/T) limit curves which would be valid to 15 effective full power years. The changes include a new fluence determination based on the surveillance capsule report WCAP-14484 and the use of the American Society of Mechanical Engineers (ASME) Code Case N-626. That Code Case has been renamed to ASME Code Case N-640, and allows the P/T curves to be developed using the K_{IC} fracture toughness curve of ASME Section XI, Appendix A, instead of the K_{Ia} curve of Appendix G. The changes affect TS sections 3.4.9.1 and 3.4.9.3, and the associated bases. Additionally, Figure 3.4-3 is separated into Figures 3.4-3a, -3b, -3c, -3d, and -3e for clarity of the illustration.

Additionally, the June 17, 1999, letter requested an exemption from the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Section 50.60(a), and 10 CFR Part 50, Appendix G, to allow application of Code Case N-640 in establishing the reactor vessel pressure limits at low temperatures for BVPS-2. Based upon the review of the information provided, the U.S. Nuclear Regulatory Commission (NRC) staff has determined that application of Code Case N-640 to BVPS-2 is acceptable. Accordingly, the NRC, pursuant to 10 CFR 50.12(a), has issued an exemption for BVPS-2, which is also enclosed.

A copy of the related safety evaluation is also enclosed. A copy of the exemption and the Notice of Issuance of the amendment are 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

Enclosures: 1. Amendment No. 113 to NPF-73
2. Safety Evaluation
3. Exemption

cc w/encls: See next page

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* SEs dated 9/29/99, 11/30/99, 3/10/00, and 3/29/00 were provided. No major technical changes were made.

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PENNSYLVANIA POWER COMPANY

OHIO EDISON COMPANY

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

THE TOLEDO EDISON COMPANY

FIRSTENERGY NUCLEAR OPERATING COMPANY

DOCKET NO. 50-412

BEAVER VALLEY POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 113
License No. NPF-73

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by FirstEnergy Nuclear Operating Company, et al. (the licensee) dated June 17, 1999, as supplemented September 15, 1999, and February 15 and June 29, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-73 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 113, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. FENOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Marsha Gamberoni, Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 6, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 113

FACILITY OPERATING LICENSE NO. NPF-73

DOCKET NO. 50-412

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
XI	XI
3/4 4-31	3/4 4-31
3/4 4-32	3/4 4-32
- - - -	3/4 4-32a
- - - -	3/4 4-32b
- - - -	3/4 4-32c
- - - -	3/4 4-32d
3/4 4-37	3/4 4-37
B 3/4 4-7	B 3/4 4-7
B 3/4 4-8	B 3/4 4-8
B 3/4 4-9	- - - -
B 3/4 4-10	B 3/4 4-10
B 3/4 4-11	B 3/4 4-11
- - - -	B 3/4 4-11a
B 3/4 4-12	B 3/4 4-12
B 3/4 4-13	B 3/4 4-13
B 3/4 4-14	B 3/4 4-14
B 3/4 4-14a	B 3/4 4-14a
B 3/4 4-15	B 3/4 4-15
B 3/4 4-15a	B 3/4 4-15a
B 3/4 4-15b	B 3/4 4-15b
B 3/4 4-15c	B 3/4 4-15c
B 3/4 4-15d	B 3/4 4-15d
B 3/4 4-15e	B 3/4 4-15e
B 3/4 4-15f	B 3/4 4-15f
B 3/4 4-15g	B 3/4 4-15g
B 3/4 4-15h	B 3/4 4-15h
B 3/4 4-15i	B 3/4 4-15i
- - - -	B 3/4 4-15j

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 113 TO FACILITY OPERATING LICENSE NO. NPF-73

PENNSYLVANIA POWER COMPANY

OHIO EDISON COMPANY

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

THE TOLEDO EDISON COMPANY

FIRSTENERGY NUCLEAR OPERATING COMPANY

BEAVER VALLEY POWER STATION, UNIT 2

DOCKET NO. 50-412

1.0 INTRODUCTION

By letter dated June 17, 1999, Duquesne Light Company (DLC) requested a license amendment to the Beaver Valley Power Station, Unit 2 (BVPS-2) Technical Specifications (TSs). The requested amendment would approve new low temperature overpressure protection (LTOP) set points and new heatup and cooldown pressure/temperature (P/T) limit curves which would be valid to 15 effective full power years (EFPY). By letter dated September 15, 1999, DLC submitted additional clarifying information regarding the operation of the overpressure protection system.

At the time of the June 17, and September 15, 1999, letters, DLC was the licensed operator for Beaver Valley Power Station, Unit No.1 (BVPS-1) and BVPS-2. On December 3, 1999, DLC's ownership interests in both BVPS-1 and BVPS-2 were transferred to the Pennsylvania Power Company, and DLC's operating authority for BVPS-1 and BVPS-2 was transferred to FirstEnergy Nuclear Operating Company (FENOC). By letter dated December 13, 1999, FENOC requested that the U.S. Nuclear Regulatory Commission (NRC) continue to review and act upon all requests before the Commission which had been submitted by DLC. By letter dated February 15, 2000, FENOC submitted revised proprietary and nonproprietary versions of the Westinghouse reports submitted with the June 17, 1999, letter. The revised Westinghouse reports superseded the original reports in their entirety. By letter dated June 29, 2000, FENOC submitted revised markups of the TS pages which included clarifying information and editorial changes. Final typed TS pages were also included with this letter. The September 15, 1999, February 15, and June 29, 2000, letters did not change the initial proposed no significant hazards consideration determination or expand the amendment beyond the scope of the initial notice.

The proposed changes include a new fluence determination based on the surveillance capsule report WCAP-14484 and the use of the American Society of Mechanical Engineers (ASME) Code Case N-626, which was approved by the ASME Section XI in September 1998. That Code Case has been renamed to Code Case N-640 and allows the P/T curves to be developed using the K_{IC} fracture toughness curve of ASME Section XI committee, Appendix A, instead of the K_{Ia} curve of Appendix G. Approval of Code Case N-640 is addressed in the exemption. The new limits are based on a fluence projection to 15 EFPY of operation. The proposed changes will affect TS Sections 3.4.9.1 and 3.4.9.3, and the associated bases. The licensee proposes to separate Figure 3.4-3 into Figures 3.4-3a, -3b, -3c, -3d, and -3e for clarity of the illustration. For the calculation of the heatup and cooldown curves, a composite curve is constructed from the controlling locations and the desired heatup and cooldown rates. The P/T limits for criticality, in-service testing, and hydrostatic testing were calculated to comply with the requirements of Title 10 of the *Code of Federal Regulations*, (10 CFR) Part 50, Appendix G.

2.0 BACKGROUND

2.1 Regulatory Requirements

The NRC has established requirements in 10 CFR Part 50 to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. Appendix A, General Design Criterion (GDC) 14 requires that the reactor coolant system (RCS) boundary be designed, fabricated, erected and tested so as to have an extremely low probability of abnormal leakage, rapidly propagating failure or gross rupture. GDC 31 requires that sufficient margin be provided to assure that the reactor coolant pressure boundary behaves in a nonbrittle manner under the stresses of normal operation, maintenance, test, and accident conditions, with a very low probability of rapidly propagating fracture. Additional regulations and guidance are contained in 10 CFR 50.60, 10 CFR 50.61, 10 CFR Part 50, Appendix G; 10 CFR Part 50, Appendix H; Generic Letter (GL) 88-11; Regulatory Guide (RG) 1.99, Revision 2 (Rev. 2); GL 92-01, Revision 1; GL 92-01, Revision 1, Supplement 1; and NUREG-800, Standard Review Plan (SRP), Section 5.3.2.

Pursuant to 10 CFR 50.60 and 50.61, licensees must demonstrate that the effects of progressive embrittlement by neutron irradiation do not compromise the integrity of the reactor pressure vessel. To this end, two analyses are required: a determination of the P/T limits for normal heatup and cooldown operations; and an assessment of the vessel's ability to maintain its integrity during an emergency shutdown with cold water injection (i.e. pressurized thermal shock (PTS)). Appendices G and H to 10 CFR Part 50 are invoked by 10 CFR 50.60, while 10 CFR 50.61 is the PTS rule that requires a PTS assessment.

Appendix G to 10 CFR Part 50 specifies fracture toughness requirements for ferritic materials of the reactor coolant pressure boundary. It requires that P/T limits for the RCS be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code. Alternatives to Appendix G may be used via an exemption, granted by the Director of the Office of Nuclear Reactor Regulation. In this submittal, Code Case N-640 is used. Appendix H to 10 CFR Part 50 requires a reactor vessel materials surveillance program to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region. These changes result from exposure of these materials to neutron irradiation and changes of the thermal environment. Material specimens exposed in the surveillance capsules are removed and tested at specified time intervals to monitor changes in the fracture toughness of the material.

GL 88-11 advised licensees that the staff would use RG 1.99, Rev. 2, to review P/T limit curves. RG 1.99, Rev. 2, contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy resulting from neutron radiation. GL 92-01, Rev. 1, requested that licensees submit their reactor pressure vessel (RPV) data for their plants to the staff for review. GL 92-01, Rev. 1, Supplement 1, requested that licensees provide and assess data from sources, which were additional to those used in previous evaluations, that could affect their RPV integrity evaluations. These data are used by the staff as the basis for the staff's review of P/T limit curves and as the basis for the staff's review of PTS assessments (10 CFR 50.61 assessments).

SRP Section 5.3.2 provides an acceptable method of determining the P/T limit curves for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics methodology of Appendix G, to Section XI of the ASME Code. The basic parameter of this methodology is the stress intensity factor K_I, which is a function of the stress state and flaw configuration. ASME Section XI, Appendix G, requires a safety factor of 2.0 on stress intensities resulting from reactor pressure during normal and transient operating conditions, and a safety factor of 1.5 for hydrostatic testing curves. The methods of ASME Section XI, Appendix G, postulate the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum stress. This flaw is postulated to have a depth that is equal to 1/4 of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. The critical locations in the RPV beltline region for calculating heatup and cooldown P/T curves are the 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively.

The Appendix G ASME Code methodology requires that licensees determine the adjusted reference temperature (ART or adjusted RT_{NDT}). The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin (M) term (i.e.: ART = Initial RT_{NDT} + ΔRT_{NDT} + M).

The ΔRT_{NDT} is a product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Rev. 2, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the chemistry factor was determined using the tables in RG 1.99, Rev. 2, or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel contents, the fluence and the calculational procedures. RG 1.99, Rev. 2, describes the methodology to be used in calculating the margin term.

LTOP system requirements are summarized in Reactor Systems Branch Technical Position 5-2 described in the SRP.

2.2 LTOP System

The LTOP system is designed to protect the pressure vessel boundary from low temperature overpressurization by designating P/T limits which satisfy the requirements of 10 CFR Part 50, Appendix G. The material radiation embrittlement is accounted for in the calculation of the ART as discussed above. The fluence value for the application of the above formula was estimated in Section 6 of WCAP-14484.

For BVPS-2, overpressure mitigation is accomplished using a combination of pressurizer power-operated relief valves (PORVs) and residual heat removal system (RHR) relief valves. The system is manually enabled by the operator and uses staggered setpoints for the lift pressure of the PORVs. The maximum pressure is determined from a number of assumed transients including mass as well as heat addition. The results of the transient analyses indicate that the mass addition transient is limiting for RCS indicated temperatures up to 137 °F . The heat addition transient is limiting above 137 °F. Because Code Case N-640 is used, the level of the stress limit to be utilized is 100 percent of that allowed by 10 CFR Part 50, Appendix G.

3.0 EVALUATION

BVPS-2 is currently licensed for operation to 10 EFPY. The proposed revised operating limits are for 15 EFPY and are based on: (1) revised fluence values derived from the measurement and evaluation of surveillance capsule V described in WCAP-14484, (2) the most limiting transient (mass or heat injection transients), and (3) the use of Code Case N-640.

3.1 Pressure Vessel Fluence

BVPS-2 has six dosimetry surveillance capsules installed. One of them was removed at the end of cycle 1 (Capsule U), and another was removed at the end of cycle 5 (Capsule V). The 15 EFPY value was extrapolated from the estimated fluence values from both capsules. The basic fluence calculations were carried out according to staff recommendations that are contained in Draft Regulatory Guide DRG-1053, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” which is publicly available. The measurements were performed in compliance with applicable American Society for Testing and Materials (ASTM) standards, which are referenced in WCAP-12406. Therefore, the U.S. Nuclear Regulatory Commission (NRC) staff finds the proposed fluence value for 15 EFPY to be acceptable.

3.2 Transients

The peak RCS pressure above the LTOP system setpoints during a low temperature overpressure transient (overshoot) is determined from a number of assumed transients including mass as well as heat addition. The licensee’s analysis included the necessary assumptions, which are discussed in WCAP-14040, as well as an evaluation of maximum allowable PORV setpoints, RCP seal limits, LTOP enable temperature, and RCS pressure vent size.

3.2.1 Transient Input Assumptions

- Mass and heat input events are considered. Mass addition assumes a water solid RCS and isolated RHR system. Heat addition is performed for temperatures between 70 °F and 300 °F.

- Pressure overshoot event analysis is based on pressurizer PORV open/close stroke times of 1.65 sec. and 1.0 sec. respectively.

- For overshoots, the single failure criterion is observed. Namely, one of the PORVs is assumed to have failed.

- Because Code Case N-640 (K_{IC} stress intensity factor) is used, Code Case N-514 (110 percent of Appendix G stress level) does not apply.

- Setpoints are selected such that the 15 EFPY Appendix G pressure limits at 60 °F (bolt up temperature) are not exceeded.

- Maximum allowable setpoints are calculated from steady-state, heatup, and cooldown at specified hourly rates.

- The setpoints are applicable to 30 percent tube plugging.

- The setpoints account for instrument uncertainty for pressure and temperature.

- The setpoints account for the pressure differential (ΔP) between the pressure transmitter and the beltline region critical element.

- Heat transport effects in the heat injection transient account for a 50 °F difference between the temperature sensor and the vessel.

3.2.2 Maximum Allowable PORV Setpoints

The 10 CFR Part 50, Appendix G, limit, adjusted for instrument uncertainty, combined with the RCS pressure peak from the mass injection event defines the maximum PORV setpoint. The mass injection transient is limiting up to a measured RCS temperature of 137 °F. Thereafter, the heat addition transient becomes limiting. The final maximum allowable setpoints are then defined by a curve consisting of the most limiting points from 10 CFR Part 50, Appendix G; the mass and heat addition transients; instrument uncertainty; elevation ΔP; and pump operation ΔP. The following combination of reactor coolant pump (RCP) operation will protect the Appendix G limits:

$T_{RCS} \leq 190$ °F	With 0-2 RCPs running
$T_{RCS} > 190$ °F	With 3 RCPs running

The maximum allowable nominal PORV setpoint for the LTOP system is shown in TS Figure 3.4-4. For indicated RCS temperatures up to 140 °F, the setpoint is 454 psig.

The NRC staff has reviewed the licensee's evaluation and finds that it results in conservative PORV setpoints that will protect the RCS from exceeding the limits of 10 CFR Part 50, Appendix G, during mass and heat addition transients. Therefore the maximum allowable nominal PORV setpoints for the LTOP system are acceptable.

3.2.3 Reactor Coolant Pump Seal Limit

During a mass or heat addition transient, should two PORVs actuate simultaneously, it is estimated that the pressure undershoot will be greater than the allowable ΔP across the number one RCP seal. For this reason, the PORV settings are staggered. However, it is possible for an RHR relief valve and a pressurizer PORV to be activated simultaneously resulting in a pressure undershoot. For this reason, the RHR system is isolated from the RCS when the overpressure protection system is armed. The above results are from the licensee's design basis calculation, which is the analysis of record. The NRC staff finds this to be acceptable since it provides adequate protection for the number one RCP seal.

3.2.4 Enable Temperature

The initial submittal did not include a discussion of the enable temperature. Additional information submitted in DLC's September 15, 1999, letter clarified that the enable temperature remained at 350 °F. However, the licensee's supplemental submittal dated February 15, 2000, indicated a revised administrative LTOP enable temperature of 367 °F. The enable temperature could have been lowered, however, the licensee opted not to do so. The higher LTOP enable temperature is conservative and, therefore, acceptable.

3.2.5 RCS Passive Vent

The RCS has a 3.14 square inch passive vent that is capable of mitigating the assumed overpressure transient. The flow capacity of the vent is greater than the limiting transient of the overpresssure protection system; with the operable charging actuated, the RCS pressure will remain bellow the P/T limit curve. Therefore, the size of the passive RCS vent is acceptable.

3.3 Pressure/Temperature Limit Methodology and Use of Code Case N-640

3.3.1 Licensee Evaluation

The licensee submitted ART calculations and P/T limit curves valid for up to 15 EFPYs. For the BVPS-2 reactor vessel, the licensee determined that the most limiting material at the 1/4T and 3/4T locations is the intermediate shell plate B9004-1 that was fabricated using plate heat number C0544-1. The licensee calculated an ART of 140 °F at the 1/4T location and 128 °F at the 3/4T location at 15 EFPYs. The neutron fluence used in the ART calculation is 1.13 X 10^{19} n/cm^2 at the 1/4T location and 0.44 X 10^{19} n/cm^2 at the 3/4T location. The ΔRT_{NDT} values at the 1/4T and 3/4T locations are 45.5 °F and 33.9 °F, respectively. The initial RT_{NDT} for the limiting plate is 60 °F. The margin term used in calculating the ART for the limiting plate is 34 °F at the 1/4T and 3/4T locations, as permitted by Position 1.1 of RG 1.99, Revision 2.

3.3.2 Staff Evaluation

The proposed P/T limits were developed by Westinghouse and are documented in WCAP-15139. This report is an update of the current P/T curves to 15 EFPY, including application of Code Case N-640. The requirements of 10 CFR Part 50, Appendix G, are also satisfied. The method in determining the P/T limits examines the beltline region, the closure head, and the nozzle region for normal heatup and cooldown, and in-service leak and hydrostatic tests.

Code Case N-640 allows the use of the reference stress intensity factor K_{IC}, which is determined by the metal temperature at the tip of a postulated flaw at the 1/4T and 3/4T locations of the vessel thickness. Appendix G of 10 CFR Part 50 defines a reference flaw in the inside vessel controlling location.

During cooldown the thermal gradients produce tensile thermal stresses which increase with increasing cooldown rate. Thus, allowable cooldown P/T relations are developed including steady-state and finite rates. From these functional relations a composite limit curve is constructed. A similar procedure is followed for the heatup curves with the difference that the 3/4T location is examined.

Appendix G requires that the temperature of the head flange and closure flange regions exceed the unirradiated RT_{NDT} by at least 120 °F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure. This value is 621 psig for BVPS-2. The limiting unirradiated RT_{NDT} occurs in the vessel flange and the minimum allowable temperature is 130 °F including 10 °F for instrument uncertainty. Instrument uncertainties were estimated following plant procedures and are applied to all P/T relations as: 10 °F, and 60 psig.

The NRC staff performed an independent calculation of the ART values for the limiting material using the methodology in RG 1.99, Revision 2. Based on these calculations, the staff verified that the licensee's limiting material for the BVPS-2 reactor vessel is intermediate shell plate B9004-1. The NRC staff's calculated ART value for the limiting material agreed with the licensee's calculated ART value.

The NRC staff evaluated the licensee’s P/T limit curves for acceptability by performing independent calculations, using the methodology referenced in the ASME Boiler and Pressure Vessel Code (as indicated by SRP 5.3.2), and verified that the licensee’s proposed P/T limits satisfy the requirements in paragraph IV.A.2 of 10 CFR Part 50, Appendix G. In addition, the staff independently generated P/T limit curves for normal operations and hydrostatic test pressures effective to 15 EFPY for BVPS-2. By comparing the independently generated P/T curves with the licensee’s curves, the NRC staff determined that the licensee’s proposed P/T limit curves meet the requirements of Appendix G of Section XI of the ASME Code, as modified by Code Case N-640. As addressed in the Exemption which is Enclosure 3 to the cover letter of this safety evaluation, the NRC staff finds that application of Code Case N-640 is acceptable. Therefore, the staff determined that the licensee’s proposed P/T limit curves are acceptable because they meet the requirements of 10 CFR 50.60 and 10 CFR Part 50, Appendix G.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes a minimum temperature at the closure head flange based on the reference temperature for the flange material. Section IV.A.2 of Appendix G states that when the pressure exceeds 20 percent 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 RT_{NDT} of -10 °F for BVPS-2, the staff has determined that the proposed P/T limits have satisfied the requirement for the closure flange region during normal operation and inservice leak and hydrostatic testing.

3.4 Technical Specification Changes

As stated previously, TS sections 3.4.9.1 and 3.4.9.3 are affected. The P/T limits in Figure 3.4-2 have been modified to reflect the 15 EFPY changes. Figure 3.4-3 is replaced with Figures 3.4-3a, -3b, -3c, -3d, and -3e, representing cooldown rates of: 0 °F, 20 °F, 40 °F, 60 °F and 100 °F respectively. This will add to the clarity of the figures. The PORV setpoint is presented in the revised Figure 3.4-4. These Figure changes represent corresponding changes in the text of the TSs.

The NRC staff finds that the TS changes represent the required adjustments for the proposed operation to 15 EFPY and are, therefore, acceptable.

3.5 Summary

The NRC staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, hydrotest, and criticality satisfy the requirements in Appendix G to Section XI of the ASME Code, as modified by Code Case N-640, and Appendix G of 10 CFR Part 50 for 15 EFPYs. The proposed P/T limits also satisfy GL 88-11, because the method in RG 1.99, Rev. 2, was used to calculate the ART. Hence, the proposed P/T limit curves may be incorporated into the BVPS-2 TSs.

4.0 STATE CONSULTATION

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

5.0 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 (64 FR 62707). 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.

6.0 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.

7.0 REFERENCES

1. WCAP-12406, "Analysis of Capsule U from Duquesne Light Company Beaver Valley Unit 2 Reactor Vessel Radiation Surveillance Program," S. E. Yanichko, et al., September 1989.

2. WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," J. D. Andracheck et al., January 1996.

3. WCAP-14484, "Analysis of Capsule V from Duquesne Light Company Beaver Valley Unit 2 Reactor Vessel Radiation Surveillance Program," P.A. Grendys, et al., September 1985.

4. WCAP-15139, "Beaver Valley Unit 2 Heatup and Cooldown Curves for Normal Operation at 15 EFPY using Code Case N-626," T. Laubham, January 1999.

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Date: September 6, 2000

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
FIRSTENERGY NUCLEAR OPERATING COMPANY	)	Docket No. 50-412
	)	
(Beaver Valley Power Station, Unit 2)	)	

EXEMPTION

I.

The FirstEnergy Nuclear Operating Company (FENOC/the licensee) is the holder of Facility Operating License No. NPF-73 that authorizes operation of the Beaver Valley Power Station, Unit 2. The license provides, among other things, that the licensee is subject to all rules, regulations, and orders of the U.S. Nuclear Regulatory Commission (NRC, the Commission) now or hereafter in effect.

The facility consists of a pressurized water reactor located in Shippingport, Beaver County, Pennsylvania.

II.

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Section 50.60(a), requires that "all light-water nuclear power reactors ... must meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in appendices G and H to this part." Appendix G to 10 CFR Part 50, requires that pressure-temperature (P/T) limits be established for reactor pressure vessels (RPVs) during normal operating and hydrostatic or leak rate testing conditions. Specifically, this regulation states that "[t]he appropriate requirements on...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 the American Society of Mechanical Engineers (ASME) Code, Section XI, Appendix G, Limits. This section of the ASME Code in turn specifies that RPV P/T limits be developed using the K_{Ia} fracture toughness curve of ASME Section XI, Appendix G, Figure G-2210-1, as the lower bound for fracture toughness.

Pressurized water reactor licensees have installed low temperature overpressure protection (LTOP) systems in order to protect the reactor coolant pressure boundary (RCPB) from being operated outside of the boundaries established by the P/T limit curves and to provide pressure relief of the RCPB during low temperature overpressurization events. The licensee is required by the Beaver Valley Unit 2 Technical Specifications (TSs) to update and submit the changes to its LTOP setpoints whenever the licensee is requesting approval for amendments to the P/T limit curves in the Beaver Valley Unit 2 TSs.

In order to address provisions of amendments to the TS P/T limits and LTOP curves, the licensee requested in its submittal dated June 17, 1999, that the staff exempt Beaver Valley Unit 2 from application of specific requirements of 10 CFR Part 50, Section 50.60(a), and 10 CFR Part 50, Appendix G, and substitute the use of ASME Code Case N-640. It should be noted that, as a result of ASME Code committee action, the original designation for this Code Case (N-626) was changed to N-640. Therefore, Code Case N-640 will be discussed below rather than Code Case N-626, which is the designation referenced in Attachments C and D of the submittal. Code Case N-640 is an alternate reference for fracture toughness for reactor vessel materials for use in determining the P/T limits.

The proposed action is in accordance with the licensee's application for exemption contained in a submittal dated June 17, 1999, and is needed to support the TS amendment that is contained in the same submittal. The proposed amendment will revise the P/T limits of TS 3/4.4.9 for Beaver Valley Unit 2 related to the heatup, cooldown, and inservice test limitations for the reactor coolant system (RCS) to 15 Effective Full Power Years (EFPYs). It will also revise the section of the TSs that relates to the overpressure protection system (OPPS) to reflect the revised P/T limits of the reactor vessels.

Code Case N-640 (formerly Code Case N-626)

The licensee has proposed an exemption to allow the use of ASME Code Case N-640 in conjunction with ASME Section XI, 10 CFR 50.60(a), and 10 CFR Part 50, Appendix G.

The proposed amendment to revise the P/T limits for Beaver Valley Unit 2, relies, in part, on the requested exemption. In accordance with Code Case N-640, these revised P/T limits have been developed using the K_{IC} fracture toughness curve shown in ASME Section XI, Appendix A, Figure A-2200-1, in lieu of the K_{Ia} fracture toughness curve of ASME Section XI, Appendix G, Figure G-2210-1, as the lower bound for fracture toughness. The other margins involved with the ASME Section XI, Appendix G, process of determining P/T limit curves remain unchanged.

Use of the K_{IC} curve in determining the lower bound fracture toughness in the development of the P/T operating limits curve is more technically correct than the K_{Ia} curve. The K_{IC} curve appropriately implements the use of static initiation fracture toughness behavior to evaluate the controlled heatup and cooldown process of a reactor vessel. The use of the initial conservatism of the K_{Ia} curve when the curve was codified in 1974 was justified. This initial conservatism was necessary due to the limited knowledge of RPV materials. Since 1974, however, additional knowledge has been gained about RPV materials, which demonstrates that the lower bound on fracture toughness provided by the K_{Ia} curve is well beyond the margin of safety required to protect the public health and safety from potential RPV failure. In addition, P/T curves based on the K_{IC} curve will enhance overall plant safety by opening the P/T operating window with the greatest safety benefit in the region of low temperature operations.

Current OPPS setpoints produce operational constraints by limiting the P/T range available to the operator for heatup or cooldown of the plant. The operating window through which the operator heats up and cools down the RCS is established by the difference between the maximum allowable pressure determined by Appendix G of ASME Section XI and the minimum required pressure for the reactor coolant pump (RCP) seals adjusted for OPPS overshoot and instrument uncertainties. The operating window becomes more restrictive with continued reactor vessel service.

Since the RCS P/T operating window is defined by the P/T operating and test limit curves developed in accordance with the ASME Section XI, Appendix G, procedure, continued operation of Beaver Valley Unit 2 with these P/T curves without the relief provided by ASME Code Case N-640 would unnecessarily restrict the P/T operating window, especially at low temperature conditions. 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 RCP start and swapping operating charging pumps with the RCS in a water-solid condition.

Additionally, the impact on the P/T limits and OPPS setpoints has been evaluated for an increased service period to 15 EFPYs based on ASME 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. Implementation of the proposed P-T curves, as allowed by ASME Code Case N-640, does not significantly reduce the margin of safety. Thus, pursuant to 10 CFR 50.12(a)(2)(ii), the underlying purpose of the regulation will continue to be served.

In summary, the ASME Section XI, Appendix G, procedure was conservatively developed based on the level of knowledge existing in 1974 concerning RPV materials and the estimated effects of operation. Since 1974, the level of knowledge about these topics has been greatly expanded. The NRC staff concurs that this increased knowledge permits relaxation of the ASME Section XI, Appendix G, requirements by application of ASME Code Case N-640, while maintaining, pursuant to 10 CFR50.12(a)(2)(ii), the underlying purpose of the ASME Code and the NRC regulations to ensure an acceptable margin of safety.

III.

Pursuant to 10 CFR 50.12, the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR Part 50, when (1) the exemptions are authorized by law, will not present an undue risk to public health or safety, and are consistent with the common defense and security; and (2) when special circumstances are present. Special circumstances are present whenever, according to 10 CFR 50.12(a)(2)(ii), "Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule...."

The underlying 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. The NRC staff accepts the licensee's determination that an exemption would be required to approve the use of Code Case N-640.

The NRC staff examined the licensee's rationale to support the exemption request. Based upon a consideration of the conservatism that is explicitly incorporated into the methodologies of 10 CFR Part 50, Appendix G; ASME Section XI, Appendix G; and Regulatory

Guide 1.99, Revision 2, the NRC staff finds that the application of Code Case N-640 will provide results which are sufficiently conservative to ensure the integrity of the reactor coolant pressure boundary and, thus, meet the underlying intent of 10 CFR Part 50, Section 50.60(a), and 10 CFR Part 50, Appendix G. This is also consistent with determinations that the NRC staff has reached for other licensees under similar conditions, and based on the same considerations. Therefore, the NRC staff finds that special circumstances set forth in 10 CFR 50.12(a)(2)(ii) are present and that the methodology of Code Case N-640 may be used to revise the P/T limits and the LTOP setpoints for the Beaver Valley Unit 2 RCS.

IV.

Accordingly, the Commission has determined that, pursuant to 10 CFR 50.12(a), the exemption is authorized by law, will not endanger life or property or common defense and security, and is, otherwise, in the public interest. Therefore, the Commission hereby grants FENOC an exemption from the requirements of 10 CFR Part 50, Section 50.60(a), and 10 CFR Part 50, Appendix G, for the Beaver Valley Unit 2 reactor coolant system.

Pursuant to 10 CFR 51.32, the Commission has determined that the granting of this exemption will not result in any significant effect on the quality of the human environment. (65 FR 50722).

This exemption is effective upon issuance.

Dated at Rockville, Maryland, this 6 day of September 2000.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

John A. Zwolinski, Director
Division of Licensing Project Management
Office of Nuclear Reactor Regulation