February 20, 2002

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

SUBJECT: BEAVER VALLEY POWER STATION, UNIT 1 - ISSUANCE OF AMENDMENT RE: AMENDED PRESSURE-TEMPERATURE LIMITS (TAC NO. MB2301)

Dear Mr. Myers:

The Commission has issued Amendment No. 249 to Facility Operating License No. DPR-66 for the Beaver Valley Power Station, Unit 1 (BVPS-1). This amendment consists of changes to the BVPS-1 Technical Specifications in response to your application dated June 29, 2001, as supplemented by letters dated October 4 and December 1, 2001. The amendment revises the pressure-temperature curves and the cold overpressure protection limits. The changes are based on a new fluence determination based on evaluation of a surveillance capsule, and the use of the American Society of Mechanical Engineers (ASME) Code Case N-640.

Your June 29, 2001, letter also requested 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, to allow application of ASME Code Case N-640 in establishing the reactor vessel pressure limits at low temperatures. The requested exemption has been issued separately.

A copy of the related safety evaluation supporting the amendment is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

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

Docket No. 50-334

Enclosures: 1. Amendment No. 249 to DPR-66
2. Safety Evaluation

cc w/encls: See next page

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cc w/encls: See next page
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Beaver Valley Power Station, Units 1 and 2

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PENNSYLVANIA POWER COMPANY

OHIO EDISON COMPANY

FIRSTENERGY NUCLEAR OPERATING COMPANY

DOCKET NO. 50-334

BEAVER VALLEY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 249
License No. DPR-66

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 29, 2001, as supplemented by letters dated October 4 and December 1, 2001, 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. DPR-66 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 249, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Joel T. Munday, Acting 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: February 20, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 249

FACILITY OPERATING LICENSE NO. DPR-66

DOCKET NO. 50-334

Replace the following pages of 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
XIX	XIX
3/4 4-2c	3/4 4-2c
3/4 4-24	3/4 4-24
3/2 4-25	3/4 4-25
3/4 4-27a	3/2 4-27a

The Technical Specification Bases sections of the TS are controlled by the licensee under TS Section 6.18, "Technical Specifications (TS) Bases Control Program." The NRC staff recognizes that the licensee will issue retyped pages to reflect the changes indicated in the licensee's amendment application. These pages are:

B 3/4 4-1
B 3/4 4-5
B 3/4 4-6
B 3/4 4-6a
B 3/4 4-6b
B 3/4 4-7
B 3/4 4-7a
B 3/4 4-7b
B 3/4 4-8
B 3/4 4-8a
B 3/4 4-9
B 3/4 4-10
B 3/4 4-10a
B 3/4 4-10c
B 3/4 4-10d
B 3/4 4-10e
B 3/4 4-10f

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 249 TO FACILITY OPERATING LICENSE NO. DPR-66

PENNSYLVANIA POWER COMPANY

OHIO EDISON COMPANY

FIRSTENERGY NUCLEAR OPERATING COMPANY

BEAVER VALLEY POWER STATION, UNIT NO. 1

DOCKET NO. 50-334

1.0 INTRODUCTION

By letter dated June 29, 2001, as supplemented October 4 and December 1, 2001, FirstEnergy Nuclear Operating Company (FENOC, the licensee) submitted a request for changes to the Beaver Valley Power Station, Unit No. 1 (BVPS-1) Technical Specifications (TSs). The requested changes would revise the pressure temperature (P-T) curves and the cold overpressure protection limits (Reference 1). The calculations of the revised P-T curves are delineated in WCAP-15570, Revision 2 (Reference 2), and the calculations for the overpressure protection limits are in a Westinghouse report (Reference 3). The proposed changes affect TS Section 3/4.4.9.3 and the associated bases. The licensee submitted additional information on October 4, 2001 (Reference 4) and December 1, 2001 (Reference 7). The supplemental letters provided additional information but did not change the initial proposed no significant hazards consideration determination or expand the amendment beyond the scope of the initial notice.

The licensee recently removed and measured surveillance capsule Y. The licensee used the results to form the basis for revised fluence and associated material properties (Reference 5). The capsule report included updated values for capsules V, U and W. The updating consists of improved methods, use of current cross sections, and adherence to the guidance of Regulatory Guide (RG) 1.190. Because the fluence values were reevaluated, the licensee recalculated RT_{PTS} to the end of the current license at 28 effective full power years (EFPYs) of operation as required by Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.61 (Reference 6).

The licensee proposed to revise P-T limits which would be effective through 22 EFPYs of facility operation. The proposed changes to the P-T limits were based, in part, on the use of American Society of Mechanical Engineers (ASME) Code Case N-640. Since Appendix G to 10 CFR 50 mandates use of Appendix G to Section XI of the ASME Code for developing reactor pressure vessel (RPV) P-T limits, the licensee requested an exemption in order to use ASME Code Case N-640. The Commission has granted an exemption from 10 CFR 50.60(a), and 10 CFR Part 50, Appendix G. A copy of the exemption is enclosed with the letter transmitting this safety evaluation.

2.0 EVALUATION

2.1 Regulatory Requirements and Guidance

The Nuclear Regulatory Commission (NRC) has established requirements in 10 CFR Part 50 to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The NRC staff evaluates the P-T limit curves based on the following NRC regulations and guidance: Appendix G to 10 CFR Part 50; Generic Letter (GL) 88-11; GL 92-01, Revision 1; GL 92-01, Revision 1, Supplement 1; RG 1.99, Revision 2 (RG 1.99, Rev. 2); and Standard Review Plan (SRP), Section 5.3.2. 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 (USE) 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 other licensees that could affect their RPV integrity evaluations. These data are used by the staff as the basis for the review of P-T limit curves. Appendix G to 10 CFR Part 50 requires that P-T limit curves for the RPV 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, 1995 Edition through the 1996 Addenda.

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 (LEFM) 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. 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 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 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 methodology found in Appendix G to Section XI of the ASME Code 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 term.

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 (CF) 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, Revision 2, describes the methodology to be used in calculating the margin term.

2.2 Use of ASME Code Case N-640

In the license amendment request, the licensee requested, pursuant to 10 CFR 50.60(b), an exemption that would allow FENOC to deviate from complying with the requirements in 10 CFR Part 50, Appendix G, for generating the P-T limit curves. By letter dated February 19, 2002, pursuant to 10 CFR 50.12, the NRC granted an exemption to allow FENOC to deviate from the requirements o CFR Part 50, Appendix G, and to use Code Case N-640 as part of the bases for generating the BVPS-1 P-T li curves for normal operations.[1] The staff's evaluation of the proposed P-T limit curves is in part based on this exemption, and on the staff's evaluation of the RPV fast neutron fluence.

The licensee submitted ART calculations and P-T limit curves valid for up to 22 EFPY of facility operation. For the BVPS-1 RPV, the licensee determined that the most limiting material at the 1/4T location was the lower intermediate shell plate fabricated using plate heat number B6903-1 and the most limiting material at the 3/4T location was the intermediate shell plate fabricated using plate heat number B6607-2. The ART values at the 1/4T and 3/4T locations for each of these plates at 22 EFPY were 233 °F and 196 °F, respectively. The neutron fluences used in the ART calculations were 1.70 X 10^{19} n/cm^2 for plate B6903-1 at the 1/4T location and 0.662 X 10^{18} n/cm^2 for plate B6607-2 at the 3/4T location for 22 EFPY. The ΔRT_{NDT} values at 22 EFPY were 171.6 °F for plate B6903-1 at the 1/4T location based on the application of BVPS-1 plant-specific surveillance data and 88.8 °F for plate B6607-2 at the 3/4T location. Prior NRC staff evaluations have required the use of BVPS-1 plant-specific surveillance data for plate B6903-1 since the data has implied an embrittlement trend significantly higher than indicated by the generic models described by the chemistry factor tables in RG 1.99, Revision 2. The initial RT_{NDT} for plate B6903-1 was 27 °F and the initial RT_{NDT} for plate B6607-2 was 73 °F. The margin term used in calculating the ART for each plate was 34 °F (note: although the BVPS-1 plant-specific data has demonstrated a high mean embrittlement trend, it also exhibits a great deal of scatter which results in it being classified as "non-credible" and necessitates the use of the full 2 sigma margin term equal to 34 °F).

Regarding the detailed fracture mechanics evaluation performed to establish the proposed BVPS-1 P-T limits, FENOC submitted information on the throughwall temperature gradients resulting from heatup and cooldown transients and their determination of the applied stress intensity at the tip of the postulated 1/4T and 3/4T flaws due to thermal loading (i.e., K_{IT}) in an enclosure to its December 1, 2001, letter. This information, along with knowledge of the applied stress intensity at the tip of the postulated 1/4T and 3/4T flaws due to pressure loads and the material property information cited above, permitted the staff to evaluate the acceptability of the proposed BVPS-1 P-T limit curves.

Use of the K_{IC} curve in determining the lower bound fracture toughness curve in the development of P-T operating limits is more technically correct than use of 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 RPV. The staff concluded that P-T

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(1) Approval to use Code Case N-640 allows licensees to use the lower bound static initiation fracture toughness value equation (K_{Ic} equation) as the basis for establishing the P-T limits in lieu of using the lower bound crack arrest fracture toughness value equation (K_{Ia} equation), which is the method invoked by Appendix G to the Code. The NRC staff's bases for approving use of Code Case N-640 were given in an exemption dated February 19, 2002.

curves based on the K_{IC} fracture toughness curve referenced by ASME Code Case N-640 will enhance overall plant safety by opening the P-T operating window with the greatest safety benefit in the region of low-temperature operation. In addition, implementation of the proposed P-T curves, as allowed by ASME Code Case N-640, does not significantly reduce the margin of safety.

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 NRC staff verified that the licensee's limiting material for the RPV is the lower intermediate shell plate fabricated using plate heat number B6903-1. The NRC staff's calculated ART value for the limiting material agreed with the licensee's calculated ART value.

The staff evaluated the licensee's P-T limit curves for acceptability by performing a finite set of check calculations using the methodology referenced in the ASME Code (as indicated by SRP 5.3.2) based on information submitted by the licensee. Further, the staff compared information submitted by the licensee (particularly information related to the evaluation of thermal loading conditions) to information submitted previously for other, similar RPVs and determined that the information submitted by FENOC for BVPS-1 appeared to be consistent. The staff verified that the licensee's proposed P-T limits satisfy the requirements in Paragraph IV.A.2 of Appendix G of 10 CFR Part 50. Specifically, the staff concluded that the P-T limit curves submitted by the licensee were as conservative as those which would be generated by the staff's application of the methodology specified in Appendix G to Section XI of the ASME Code, as modified by ASME Code Case N-640. Therefore, the NRC staff determined that the licensee's proposed P-T limit curves were acceptable for operation of the BVPS-1 RPV though 22 EFPY of operation.

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% 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 limiting flange RT_{NDT} of 60 °F for BVPS-1, the NRC 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.

2.3 RPV Fluence and Pressurized Thermal Shock

Fluence values were calculated by the licensee and provided in the Y capsule report. In addition to the calculated values, the Y capsule report included best-estimate values calculated with FERRET, which is a non-reviewed nor approved code. However, the licensee states (Reference 2) that only calculated values were used in the estimation of the P-T curves and the overpressure protection limits. This is acceptable because the calculation method complies with the guidance in RG 1.190.

The calculational procedure described in 10 CFR 50.61 (regarding protection against pressurized thermal shock events) was used to calculate the end-of-license (28 EFPYs) parameters. The critical element is the lower shell plate B6903-1 with an RT_{PTS} value of 259 °F.

The calculated fluence value at 28 EFPY of 3.54×10^{19} n/cm^2 was used. The licensee's calculation of the chemistry factor conforms to the requirements of 10 CFR 50.61, and the value of 259 °F is lower than the screening criterion of 270 °F; therefore, it is acceptable.

2.4 Proposed TS Changes

The following TS pages are affected by the proposed amendment. The technical changes are highlighted below. Associated with these technical changes are administrative and editorial changes, which are not described here.

> TS 3.4.1.3.a.1, 2 and 3, regarding operable coolant loops -- The footnote # is revised to read: "The first reactor coolant pump in a non-isolated loop shall not be started with one or more non-isolated RCS cold leg temperatures less than or equal to the enable temperature set forth in Specification 3.4.9.3, unless the secondary side water temperature of each steam generator in a non-isolated loop is less than 50 °F [changed from the previous 25 °F.......]" The change is the result of heat input analysis as described in WCAP-14040, which was previously approved by the NRC staff.
>
> Figure 3.4-2 and 3.4-3 -- The heatup and cooldown curves are revised to be valid for 22 EFPYs, and to reflect calculation using new fluence values and ASME Code Case N-640 (Reference 3).
>
> TS 3.4.9.3, regarding overpressure protection system limit settings -- The power operated relief valve (PORV) lift setting is changed to 403 psig from 432 psig, while the overpressure protection system enable temperature is changed to 343 °F from 329 °F. For pump swap operations, two charging pumps may be operable (for injecting into the RCS) for up to one hour (changed from the previous 15 minutes). The Technical Specification Traveler Forms (TSTF) 285 has approved this change. Therefore, increase of two charging pump operability overlap up to one hour is acceptable.

The licensee proposed to revise corresponding TS Bases pages to reflect the above changes. Figure B 3/4 4-1 will be deleted because it no longer represents the vessel fluence variation as a function of EFPYs. Figure B 3/4 4-6b will be replaced with Figure B 3/4 4-1 which depicts the calculated decrease of upper shelf energy (USE) as a function of peak vessel fluence. Table B 3/4.4-1 will be updated to reflect the latest chemistry values of the vessel components. Administrative and editorial changes will also be made to reflect the use of ASME Code Case N-640 and WCAP-15570 (Reference 2). Figure B 3/4 4-3 will be renumbered to B 3/4 4-2. The TS Bases sections of the TSs are controlled by the licensee under TS Section 6.18, "Technical Specifications (TS) Bases Control Program." The NRC staff recognizes that the licensee will issue retyped pages to reflect the changes indicated in the licensee's amendment application.

The above TS and TS Bases changes reflect the results of the analyses and are acceptable.

2.5 Summary of NRC Staff Review

The NRC staff finds that the fluence used by the licensee was calculated using results of the evaluation of surveillance capsule Y, and is thus acceptable. The proposed TS and TS Bases changes reflect the ASME Code Case N-640 methodology and the NRC staff-approved methodology in WCAP-14040; these are, therefore, acceptable. Finally, the NRC staff notes that the results of the calculations are correctly reflected in the actual TS and TS Bases changes.

3.0 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.

4.0 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 (66 FR 52801). 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.

5.0 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.

6.0 REFERENCES

1. Letter from M. P. Pearson for L. W. Myers of FENOC to U.S. NRC, “Beaver Valley Power Station, Unit No. 1, Docket No. 50-334, License No. DPR-66, License Amendment Request No. 292," dated June 29, 2001.

2. WCAP-15570, Revision 2, “Beaver Valley Unit 1 Heatup and Cooldown Limit Curves for Normal Operation,” T. J. Laubham, Westinghouse Electric Company, LLC, April 2001.

3. “Beaver Valley Unit 1, FirstEnergy Nuclear Operating Company, Overpressure Protection System Setpoints for Y-Capsule,” Revision 1, by R. Calvo, Westinghouse Electric Company, LLC, April 2001.

4. Letter from L. W. Myers of FENOC to U.S. NRC, “Beaver Valley Power Station, Unit No. 1 BV-1, Docket No. 50-334, License No. DPR-66 Response to a Request for Additional Information In Support of LAR No. 292," dated October 4, 2001.

5. WCAP-15571, “Analysis of Capsule Y from Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program,” C. Brown et al., Westinghouse Electric Company, LLC, November 2000.

6. WCAP-15569, “Evaluation of Pressurized Thermal Shock for Beaver Valley Unit 1," by C. Brown and E. Terek, Westinghouse Electric Company, LLC, November 2000.

7. Letter from L. W. Myers of FENOC to U.S. NRC, “Beaver Valley Power Station, Unit No. 1 BV-1, Docket No. 50-334, License No. DPR-66 Response to a Request for Additional Information In Support of LAR No. 292," dated December 1, 2001.

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Date: February 20, 2002