April 29, 2003

Mr. William R. Kanda
Vice President - Nuclear, Perry
FirstEnergy Nuclear Operating Company
Perry Nuclear Power Plant
P.O. Box 97, A200
10 Center Road
Perry, OH 44081

SUBJECT: PERRY NUCLEAR POWER PLANT, UNIT 1 - ISSUANCE OF AMENDMENT (TAC NO. MB6061)

Dear Mr. Kanda:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 127 to Facility Operating License No. NPF-58 for the Perry Nuclear Power Plant (PNPP), Unit 1. This amendment revises the Technical Specifications in response to your application dated June 4, 2002.

This amendment revises the pressure temperature (P-T) limit curves for 9 and 18-effective full power years (EFPY) of operation to operation with P-T limits calculated for 22 and 32-EFPY for PNPP. The June 2002 application also contained a request for exemption from applying Appendix G of the 1995 American Society of Mechanical Engineers Boiler and Pressure Vessel Code and approval for using Code Case N-640, which permits the use of the plane strain fracture toughness (K_{IC}) curve instead of the crack arrest fracture toughness (K_{Ia}) curve for reactor pressure vessel materials in determining the P-T limits. The exemption is granted.

A copy of the Safety Evaluation and the exemption are also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Stephen P. Sands, Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-440

Enclosures: 1. Amendment No. 127 to License No. NPF-58
2. Safety Evaluation
3. Exemption

cc w/encls: See next page

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FIRSTENERGY NUCLEAR OPERATING COMPANY

DOCKET NO. 50-440

PERRY NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 127
License No. NPF-58

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

   A. The application for amendment by the FirstEnergy Nuclear Operating Company (the licensee) dated June 4, 2002, 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-58 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. are hereby incorporated into this license. The FirstEnergy Nuclear Operating Company 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 90 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 29, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 127

FACILITY OPERATING LICENSE NO. NPF-58

DOCKET NO. 50-440

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
3.4-31	3.4-31
3.4-31a	3.4-31a
3.4-31b	3.4-31b
3.4-31c	3.4-31c
3.4-31d	3.4-31d
3.4-31e	3.4-31e

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 127 TO FACILITY OPERATING LICENSE NO. NPF-58

FIRSTENERGY NUCLEAR OPERATING COMPANY

PERRY NUCLEAR POWER PLANT

DOCKET NO. 50-440

1.0 INTRODUCTION

By letter dated June 4, 2002, FirstEnergy Nuclear Operating Company, the licensee, submitted a technical specification (TS) change request to revise and replace the current reactor vessel pressure temperature (P-T) limit curves for 9 and 18-effective full power years (EFPY) of operation, with P-T limits calculated for 22-EFPY and 32-EFPY for Perry Nuclear Power Plant (PNPP). The submittal also contained a request for exemption from applying Appendix G of the 1995 Edition through the 1996 Addenda of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) and approval of using Code Case N-640, which permits the use of the plane strain fracture toughness (K_{IC}) curve instead of the crack arrest fracture toughness (K_{Ia}) curve for reactor pressure vessel (RPV) materials, in determining the P-T limits.

2.0 BACKGROUND

The licensee submitted material data and detailed methodologies for generating P-T limits for 22-EFPY and 32-EFPY for the bottom head, beltline, upper vessel and closure assembly for PNPP. The material information includes the initial RT_{NDT} values for all materials in the four geometric classifications mentioned above. For the beltline material, the licensee determined that the most limiting material for P-T curves is the vertical weld seam that was manufactured with the weld heat 627260 lot B322A27AE (BD,BF). The licensee employed the methodology in Regulatory Guide (RG) 1.99, Rev. 2, “Radiation Embrittlement of Reactor Vessel Materials,” and calculated an adjusted reference temperature (ART) of 63 °F (22-EFPYs) at the ¼T (where T is the minimum beltline shell thickness) fluence of 0.20E19 n/cm^2 for this limiting material based on a reference temperature of nil ductility transition (RT_{NDT}) value of 46 °F, an initial RT_{NDT} of -30 °F, and a margin term of 46 °F (σ_I = 0 °F and σ_Δ = 23 °F) and an ART of 78 °F (32-EFPYs) at the ¼T fluence of 0.29E19 n/cm^2 for this limiting material based on a RT_{NDT} value of 54 °F, an initial RT_{NDT} of -30 °F, and a margin term of 54 °F (σ_I = 0 °F and σ_Δ = 27 °F). The ΔRT_{NDT} value for this material was determined using the chemistry Table of RG 1.99, Rev. 2. The licensee did not perform similar calculations for the limiting upper vessel and bottom head material because these non-beltline materials only experienced insignificant fluence.

Based on the ART of 63 °F (22-EFPY) and 78 °F (32-EFPY) for the limiting beltline material and the highest initial RT_{NDT} value of 10 °F for both the upper vessel and the bottom head materials, the licensee used the methodology of Appendix G in the 1995 Edition through the 1996 Addenda of Section XI of the ASME Code, as modified by Code Case N-640, to calculate the P-T limits for the PNPP for 22-EFPY and 32-EFPY.

PNPP developed P-T curves for non-beltline (upper and lower head) regions primarily for consideration of bottom head discontinuity stresses during pressure testing. Non-beltline

curves were also developed for core not critical, and core critical operations using a heatup/cooldown rate of 100 °F per hour. Therefore the non-beltline core not critical, and core critical curves developed are bound by a temperature change of no greater than 100 °F in one hour. The non-beltline P-T curves were developed for a large BWR/6. PNPP is bound by the P-T curves developed for a large BWR/6 because the inner radius used in the analysis of a large BWR/6 is greater than PNPP's. The non-beltline curves were based on the most limiting properties. That is, the curve is shifted based on the most limiting initial RT_{NDT}.

The P-T curves for heatup and cooldown operating condition at a given EFPY apply for both ¼T and ¾T locations. When combining pressure and thermal stresses, it is usually necessary to evaluate stresses at the ¼T location and the ¾T location because the thermal stress of interest is at the ¼T location during cooldown and ¾T location during heatup. The licensee conservatively approximated the stresses by assuming the thermal gradient stress at ¼T to be tensile for both heatup and cooldown. This results in conservatism because irradiation effects cause the allowable toughness, K_{Ir}, at ¼T to be less than that at ¾T for a given metal temperature. Hence, the licensee's approximate approach for beltline heatup curves is conservative and acceptable.

Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR), General Design Criterion (GDC) 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," states that the reactor coolant boundary shall be designed with sufficient margin to assure that, when stressed under all modes of operation, the boundary behaves in a non-brittle manner. GDC-31 also states that reactor vessel design must include consideration of uncertainties in the determination of the effects of neutron irradiation. 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," quantifies the criteria of embrittlement in terms of the nil-ductility transition temperature of the vessel, RT_{PTS}, at the end of the plant's operating license. The staff issued RG 1.190, "Calculation and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," which summarizes the methods the staff finds acceptable in the determination of the reactor vessel fluence (Ref. 1).

3.0 REGULATORY EVALUATION

The U.S. 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 staff evaluates the P-T limit curves based on the following NRC regulations and guidance: 10 CFR Part 50, Appendix G; Generic Letter (GL) 88-11; GL 92-01, Revision 1; GL 92-01, Revision 1, Supplement 1; RG 1.99, Revision 2 (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 resulting from neutron radiation. GL 92-01, Rev. 1, requested that licensees submit their 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 staff's review of P-T limit curves and as the basis for the staff's review of pressurized thermal shock assessments (10 CFR 50.61 assessments). 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 Code.

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.

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 that is equal to ¼ thickness (¼T) 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 ¼T and ¾ thickness (¾T) locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively.

The ASME Appendix G Code methodology requires that licensees determine the ART or adjusted RT_{NDT}. 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. 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.

4.0 TECHNICAL EVALUATION

As mentioned above, the licensee requested an exemption to use ASME Code Case N-640 as the basis for establishing the 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 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, will provide an adequate level of safety with regard to RPV integrity.

The staff performed an independent calculation of the ART values for the limiting material using the methodology in RG 1.99, Rev. 2. Based on these calculations, the staff verified that the licensee's limiting material for PNPP was the vertical weld seam BD,BF. The staff's calculated ART values for the limiting material agreed with the licensee's calculated ART values of 63 °F and 78 °F.

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. 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 staff determined that the licensee's proposed P-T limit curves were acceptable for operation of the PNPP through 22 and 32-EFPY.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes minimum operational temperatures 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 160 °F for core critical operation, 120 °F for core not critical operation, and by 90 °F for hydrostatic pressure tests and leak tests. In addition, the most limiting requirements for operation at pressures below 20 percent of the preservice system hydrostatic test pressure would require that the flange temperature must exceed the reference temperature of the material in those regions by at least 60 °F, which defines the RPV head boltup temperature. Based on the limiting flange RT_{NDT} values for PNPP, the staff has determined that the proposed P-T limits have satisfied the requirements for the closure flange region for RPV head boltup, for core critical operation, for core not critical operation, and for inservice leak and hydrostatic testing.

The licensee contracted with General Electric (GE) for reactor vessel fluence calculations. GE used the methodology described in NEDC-32083P-A, which has been reviewed and approved by the NRC staff for application in licensing actions (Ref. 2). The licensee submitted the GE technical report GE-NE-0000-0000-8763-01a, which described the assumptions and approximations in the PNPP reactor vessel fluence calculations (Ref. 3). The staff review established that the approved methodology was applied correctly; therefore, the results are acceptable.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATIONS

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an Environmental Assessment and Finding of No Significant Impact has previously been prepared and published in the Federal Register on March 19, 2003 (68 FR 13335). Accordingly, based on the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect upon the quality of the human environment.

7.0 CONCLUSIONS

The staff reviewed the submitted information regarding reactor pressure vessel fluence calculations. The staff established that the methodology employed has been approved by the NRC and satisfies the guidance in RG 1.190. The methodology was applied correctly for the PNPP, thus, the estimated values are acceptable.

The staff also concludes that the proposed P-T limits curves for each of the pressure test, core not critical and core critical conditions 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. 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 PNPP TSs and shall be valid through 22-EFPY and 32-EFPY of operation.

The staff has concluded, based on the considerations 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

1. RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," US NRC, dated March 2001.

2. NEDC-32983P-A, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation," dated September 14, 2001.

3. GE-NE-0000-0000-8763-01a, "Pressure and Temperature Curves for FirstEnergy Corporation Using the K_{IC} Methodology Perry Unit 1," dated April 2002.

Principal Contributors: Robert Kuntz
Lambos Lois

Date: April 29, 2003

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

FIRSTENERGY NUCLEAR OPERATING COMPANY

PERRY NUCLEAR POWER PLANT

DOCKET NO. 50-440

EXEMPTION

1.0 BACKGROUND

The FirstEnergy Nuclear Operating Company (FENOC/ the licensee) is the holder of Facility Operating License No. NPF-58 which authorizes operation of Perry Nuclear Power Plant (PNPP). The license provides, among other things, that the facility 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 boiling water reactor located on FENOC's PNPP site, which is located in Lake County, Ohio.

2.0 REQUEST/ACTION

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix G 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, 10 CFR Part 50, Appendix G states that "[t]he appropriate requirements on...the pressure-temperature limits and minimum permissible temperature must be met for all conditions." Appendix G of 10 CFR Part 50 specifies that the requirements for these limits are the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI, Appendix G Limits.

To address provisions of amendments to the technical specifications P-T limits in the submittal dated June 4, 2002, the licensee requested that the staff exempt PNPP from application of specific requirements of 10 CFR Part 50, Section 50.60(a) and Appendix G, and substitute use of ASME Code Case N-640. Code Case N-640 permits the use of an alternate reference fracture toughness (K_{Ic} fracture toughness curve instead of 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 Code, Section XI, Appendix A, Figure A-2200-1 provides greater allowable fracture toughness than the corresponding K_{Ia} fracture toughness curve of ASME Code, Section XI, Appendix G, Figure G-2210-1, using the K_{ic} fracture toughness, as permitted by Code Case N-640, in establishing the P-T limits would be less conservative than the methodology currently endorsed by 10 CFR Part 50, Appendix G. Considering this, an exemption to apply the Code Case would be required by 10 CFR 50.60.

The licensee proposed to revise the P-T limits for PNPP using the K_{Ic} fracture toughness curve, in lieu of the K_{Ia} fracture toughness curve, as the lower bound for fracture toughness.

Use of the K_{Ic} curve in determining the lower bound fracture toughness in the development of P-T operating limits curve is more technically correct than the K_{Ia} curve since the rate of loading during a heatup or cooldown is slow and is more representative of a static condition than a dynamic condition. 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 staff has required use of the initial conservatism of the K_{Ia} curve since 1974 when the curve was codified. This initial conservatism was necessary due to the limited knowledge of RPV materials. Since 1974, 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 summary, the ASME Code, 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 Code Section XI, Appendix G requirements by applying the K_{Ic} fracture toughness, as permitted by Code Case N-640, because compliance with ASME Code, Section XI, Appendix G is not necessary to achieve the underlying purpose of 10 CFR 50.60 and Part 50, Appendix G.

3.0 DISCUSSION

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. The staff accepts the licensee's determination that an exemption would be required to approve the use of Code Case N-640. The staff examined the licensee's rationale to support the exemption request and concurred that the use of the Code Case N-640 would meet the underlying intent of these regulations. Based upon a consideration of the conservatism that is explicitly incorporated into the methodologies of 10 CFR Part 50, Appendix G; Appendix G of the Code; and Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Material," Revision 2, the staff concluded that compliance with ASME Code, Appendix G is not necessary to achieve the underlying purpose of 10 CFR 50.60 and 10 CFR Part 50, Appendix G because the application of Code Case N-640 as described would provide an adequate margin of safety against brittle failure of the RPV. This is also consistent with the determination that the staff has reached for other licensees under similar conditions based on the same considerations. Therefore, the staff concludes that requesting exemption under the

special circumstances of 10 CFR 50.12(a)(2)(ii) is appropriate and that the methodology of Code Case N-640 may be used to revise the P-T limits for PNPP.

4.0 CONCLUSION

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, exemption from the requirements of 10 CFR Part 50, Section 50.60(a) and 10 CFR Part 50, Appendix G, for PNPP.

Pursuant to 10 CFR 51.32, the Commission has determined that the granting of this exemption will not have a significant effect on the quality of the human environment (68 FR 13335).

This exemption is effective upon issuance.

Dated at Rockville, Maryland, this 29th day of April 2003.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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