February 6, 2003

Mr. Mark B. Bezilla
Vice President
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Post Office Box 4
Shippingport, PA 15077

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: POSITIVE MODERATOR TEMPERATURE COEFFICIENT (TAC NO. MB5302)

Dear Mr. Bezilla:

The Commission has issued the enclosed Amendment No. 251 to Facility Operating License No. DPR-66 for the Beaver Valley Power Station, Unit No. 1. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated May 31, 2002, as supplemented by letters dated July 19, and September 3, 2002. The amendment revises the TSs to allow operation with a positive moderator temperature coefficient.

A copy of the related safety evaluation 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. 251 to DPR-66
2. Safety Evaluation

cc w/encls: See next page

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OFFICE	PDI-1/PM	PDI-2/LA	DSSA/SRXB	OGC	PDI-1/SC
NAME	DCollins	MO'Brien	FAkstulewicz*	RWeisman	RLaufer
DATE	1/9/03	1/9/03	12/04/2002	2/5/03	2/6/03

OFFICIAL RECORD COPY

Beaver Valley Power Station, Units 1 and 2

Mary O'Reilly, Attorney
FirstEnergy Nuclear Operating Company
FirstEnergy Corporation
76 South Main Street
Akron, OH 44308

FirstEnergy Nuclear Operating Company
Regulatory Affairs/Performance Improvement
Larry R. Freeland, Manager
Beaver Valley Power Station
Post Office Box 4, BV-A
Shippingport, PA 15077

Commissioner James R. Lewis
West Virginia Division of Labor
749-B, Building No. 6
Capitol Complex
Charleston, WV 25305

Director, Utilities Department
Public Utilities Commission
180 East Broad Street
Columbus, OH 43266-0573

Director, Pennsylvania Emergency Management Agency
2605 Interstate Dr.
Harrisburg, PA 17110-9364

Ohio EPA-DERR
ATTN: Zack A. Clayton
Post Office Box 1049
Columbus, OH 43266-0149

Dr. Judith Johnsrud
National Energy Committee
Sierra Club
433 Orlando Avenue
State College, PA 16803

J. H. Lash, Plant Manager (BV-IPAB)
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Post Office Box 4
Shippingport, PA 15077

Rich Janati, Chief
Division of Nuclear Safety
Bureau of Radiation Protection
Deparment of Environmental Protection
Rachel Carson State Office Building
P.O. Box 8469
Harrisburg, PA 17105-8469

Mayor of the Borough of Shippingport
P O Box 3
Shippingport, PA 15077

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Resident Inspector
U.S. Nuclear Regulatory Commission
Post Office Box 298
Shippingport, PA 15077

FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
ATTN: M. P. Pearson, Director Services and Projects (BV-IPAB)
Post Office Box 4
Shippingport, PA 15077

FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Mr. B. F. Sepelak
Post Office Box 4, BV-A
Shippingport, PA 15077

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. 251
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 May 31, 2002, as supplemented by letters dated July 19, and September 3, 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. DPR-66 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 251, 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. Additionally, administrative controls shall be established and instituted, prior to the first entry into Mode 2 for Unit 1, Cycle 16 operations, to ensure that the moderator temperature coefficient at hot full power conditions will be maintained at a value less than or equal to -5.5 pcm/°F at all times during core life.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard J. Laufer, 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 6, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 251

FACILITY OPERATING LICENSE NO. DPR-66

DOCKET NO. 50-334

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 1-5	3/4 1-5
-	3/4 1-5a

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 251 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 application dated May 31, 2002, as supplemented by letters dated July 19, and September 3, 2002, the FirstEnergy Nuclear Operating Company (FENOC, the licensee), requested changes to the Technical Specifications (TSs) for Beaver Valley Power Station, Unit 1 (BVPS-1). The supplements dated July 19, and September 3, 2002, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on September 17, 2002 (67 FR 58644).

The proposed changes would allow operation with a positive moderator temperature coefficient (PMTC). Specifically, the proposed changes would revise TS 3.1.1.4.a for BVPS-1 to change the moderator temperature coefficient (MTC) from "less positive than 0×10^{-4} $\Delta k/k/^{\circ}F$" to "less positive than $+0.2 \times 10^{-4}$ $\Delta k/k/^{\circ}F$ for power levels up to 70% of rated thermal power, with a linear ramp to 0×10^{-4} $\Delta k/k/^{\circ}F$ at 100% rated thermal power." The overall change allows BVPS-1 to operate with a PMTC.

The requested amendment is similar to the changes requested by FENOC in a letter dated June 28, 2001, and approved for Beaver Valley Power Station, Unit No. 2 (BVPS-2), that were approved by the NRC staff in Amendment No. 129 to Operating License No. NPF-73, on February 21, 2002.

2.0 REGULATORY EVALUATION

A negative MTC results in negative reactivity feedback with increasing moderator temperature; conversely, a PMTC results in positive reactivity feedback with increasing moderator temperature. Thus, a negative MTC aids in controlling rapid increases in reactivity during accidents that cause the reactor coolant system (RCS) to heat up. Most licensees operate their plants with a negative MTC over a majority of each core cycle.

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 11, "Reactor Inherent Protection," states that "[t]he reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity." If the reactor core operates with a PMTC, this criterion can still be met through the compensating effects of the fuel temperature coefficient (FTC). The FTC, also called the Doppler coefficient, is a measure of the change in reactivity per degree of temperature change in the fuel. Similar to the MTC, a negative FTC means that there will be a decrease in reactivity with an increase in fuel temperature. As explained more fully below, the FTC is always negative and it is more dominant than the PMTC proposed by FENOC.

3.0 <u>TECHNICAL EVALUATION</u>

A PMTC will have unfavorable effects on accidents that result in an increase in the reactor coolant temperature. FENOC did not reanalyze accidents which resulted in reactor coolant temperature decrease. These accidents include (1) Feedwater Malfunction Resulting in Increased Flow (UFSAR 14.1.9), (2) Excessive Load Increase (UFSAR 14.1.10), (3) Steamline Break (UFSAR 14.2.5.1), (4) Dropped Rod (UFSAR 14.1.3), and (5) Spurious SI (UFSAR 14.1.16). The NRC staff agrees with FENOC's assessment that it is unnecessary to re-analyze the above accidents because the PMTC imparts a safety benefit in these scenarios.

For accidents that do result in an increase in the reactor coolant temperature, the licensee provided tables summarizing the initial conditions assumed and the results of the analyses. The NRC staff reviewed the results of the analyses and determined that the licensee used NRC-approved methodologies with conservative power level and MTC assumptions in the initial conditions for each event. The power level used in the analysis reflects a 1.4% power uprate that the NRC staff approved on September 24, 2001, through Amendment No. 243 to Operating License DPR-66 for BVPS-1, and Amendment No. 122 to Operating License NPF-73 for BVPS-2. As set forth below, the NRC staff concludes that the licensee preserves a safety margin and meets all of the regulatory requirements for the accidents affected by a PMTC.

The safety analysis for the above-mentioned accidents is most conservative at the highest power (100% RTP or greater). When MTC is plotted against temperature, the resulting curve has a relatively flat slope for the range of values near the operating conditions of interest. This means that the amount of reactivity increase that results from a temperature change is not very sensitive to the initial starting temperature. In other words, the reactivity increase due to a temperature change with a low initial temperature is not substantially different from the reactivity increase due to a temperature change with a high initial temperature. The most conservative initial temperature assumption is the highest moderator temperature, which corresponds to the highest power. That is because at a higher temperature the moderator is closer to the fuel temperature and, thus, has a lower ability to cool the fuel. Hence, at higher moderator temperatures, core temperatures will tend to be closer to fuel temperature limits. Additionally, the margin to departure from nucleate boiling (DNB) is smallest when the reactor is operating at its maximum power level. Therefore, accident analyses that assume high moderator temperature and high power conditions are the most conservative.

For most of the accident events, the licensee chose an MTC of +2 pcm/°F ($+0.2 \times 10^{-4}\ \Delta k/k/°F$) at 100% or greater rated thermal power (RTP). Because these values are bounding, the NRC staff finds these initial power and MTC conditions acceptable. In some cases, FENOC assumed initial conditions of 100% RTP with an MTC of 0 pcm/°F. This is consistent with FENOC's TS, and the staff finds this acceptable with the constraint that these conditions bound the results of accidents occurring at an MTC of +2 pcm/°F at part power.

Each accident has unique acceptance criteria. The list of the accidents and the NRC staff's assessment of each follows.

A. Rod Withdrawal from Subcritical (Updated Final Safety Analysis Report (UFSAR) 14.1.1)

FENOC assumed hot zero power (HZP) as the initial power level with an MTC of +2 pcm/°F. The licensee meets the minimum departure from nucleate boiling ratio (DNBR).

B. Rod Withdrawal at Power (UFSAR 14.1.2)

FENOC assumed 100% RTP (2697 MWt which is the sum of nominal core power of 2689 MWt and reactor coolant pump heat of 8 MWt) as the initial power level with an MTC of +2 pcm/°F. They also analyzed cases at 60% RTP and 10% RTP. The analyses illustrate that FENOC still meets the DNBR and peak secondary pressure limits. Generic Westinghouse analyses, which bound the conditions at BVPS-1, demonstrate that FENOC meets peak primary pressure limits.

C. Loss of Load/Turbine Trip (UFSAR 14.1.7)

FENOC assumed an MTC of +2 pcm/°F and two different power levels. They assumed a power level of 100% RTP for the DNB calculations and a power level of 100.6% RTP (2713.2 MWt) for peak pressure calculations. The analyses met the minimum value of the DNBR limit, and the maximum peak primary pressure and peak secondary pressure limits.

D. Loss of Normal Feedwater (UFSAR 14.1.8)

FENOC assumed 100.6% RTP as the initial power level with an MTC of 0 pcm/°F. They found that these conditions were more limiting than a PMTC at partial power. The loss of load event bounds DNBR and peak reactor coolant pressure. In addition, FENOC demonstrated that the pressurizer will not become water solid.

E. Loss of AC Power (UFSAR 14.1.11)

FENOC assumed 100.6% RTP as the initial power level with an MTC of 0 pcm/°F. They found that these conditions were more limiting than a PMTC at partial power. The loss of load event bounds this case for overpressurization; while the loss of flow event bounds it for DNBR. In addition, FENOC demonstrated that the pressurizer will not become water solid.

F. RCS Depressurization (UFSAR 14.1.15)

FENOC assumed 100% RTP as the initial power level with an MTC of +2 pcm/°F. This event meets the minimum DNBR.

G. Partial/Complete Loss of Flow (UFSAR 14.1.5/14.2.9)

FENOC assumed 100% RTP as the initial power level with an MTC of 0 pcm/°F for 100% and two partial loss of flow events. They found that these conditions were more limiting than a PMTC at partial power. They demonstrated that the partial and complete loss of flow events will meet the DNBR, peak reactor coolant pressure, and peak secondary pressure limits.

H. Locked Rotor (UFSAR 14.2.7)

FENOC assumed an MTC of 0 pcm/°F and two different power levels. They assumed a power level of 100% RTP for the DNB calculations and a power level of 100.6% RTP for peak pressure calculations. They found that these conditions were more limiting than a PMTC at partial power. Their design basis remains valid. They meet the peak reactor coolant pressure limits and the core will remain capable of maintaining a coolable geometry.

I. Rod Ejection (UFSAR 14.2.6)

FENOC assumed initial conditions that consist of combinations of 100.6% nominal core power (2705 MWt) and HZP at beginning of life and end of life times in order to bound the fuel cycle and expected operating conditions. This analysis models an isothermal temperature coefficient which bounds the PMTC. The licensee showed that, in all four cases, safety limits for fuel damage are not exceeded.

J. Feedline Break (UFSAR 14.2.5.2)

FENOC assumed 100.6% RTP as the initial power level with an MTC of +2 pcm/°F. The loss of load transient bounds this event for excess pressure conditions. FENOC demonstrated that there was sufficient margin to the hot leg boiling to preclude loss of coolable geometry.

3.1 Control Systems Margin to Trip Evaluation

In response to the staff's request, the licensee, in their letter dated September 3, 2002, provided detailed results of their analyses for the following most limiting Condition I transients: 50% load rejection from 100% power; 10% step load increase from 90% power; 5% per minute ramp load increase from 15% to 100% power; and, turbine trip without reactor trip from P-9 setpoint. The results of the analyses confirm that there are no challenges to the reactor trip or engineered safety feature actuation system (ESFAS) during the Condition I operating transients. The results of these analyses show that sufficient margin exists for preventing an inadvertent reactor trip during any of the limiting Condition I transients associated for a core designed with a PMTC at BVPS-1. Accordingly, we find the proposed amendment acceptable with respect to such transients.

3.2 Anticipated Transients without Scram (ATWS)

FENOC evaluated the impact of a PMTC on ATWS events. Since BVPS-1 is a Westinghouse plant, under 10 CFR 50.62, FENOC is required to have equipment, diverse from the reactor trip system from sensor output to final actuation device, to automatically initiate the auxiliary (or emergency) feedwater system and a turbine trip under conditions indicative of an ATWS. The system at BVPS-1 was reviewed and approved by the NRC on May 31, 1988. The NRC's reason for this requirement is to preclude the plant from exceeding the American Society of Mechanical Engineers Stress Level C Limit of 3200 psig in the RCS during an ATWS.

In a letter dated October 3, 1983, Westinghouse provided information on the effects of MTC on peak pressure during ATWS events. They showed that a generic four-loop plant will not exceed the 3200 psig limit at hot full power (HFP) if the MTC is more negative than -5.5 pcm/°F and if no power-operated relief valves (PORV) were blocked. Blocking a PORV will reduce the amount of pressure relieved in a plant and, therefore, make ATWS events more difficult to mitigate. The generic analyses show that for ATWS events the four-loop design is more limiting than the three-loop design because of its higher rated thermal power. Since the MTC limit of -5.5 pcm/°F applies to a Westinghouse four-loop plant, it is bounding for BVPS-1, a three-loop plant.

FENOC has guaranteed to enforce limits that will maintain an MTC more negative than -5.5 pcm/°F at HFP conditions during all times in core life. FENOC has verified to the staff that any blocked PORVs will be opened. Since peak pressure is not to exceed 3200 psig under these conditions, FENOC has correspondingly committed to meeting a 0% unfavorable exposure time.

The -5.5 pcm/°F MTC value will be reflected as a limit for the BVPS-1 reload safety analysis checklist (RSAC) which is utilized as part of the NRC-approved Westinghouse Reload Safety Evaluation Methodology. The value will be set as a reload design constraint in the RSAC for BVPS-1. FENOC's proposed limitation shows BVPS-1's ability to meet all ATWS requirements set forth by the NRC. Committing to the constraints of the four loop Westinghouse plant, FENOC is meeting the more restrictive and conservative limits despite their three-loop Westinghouse reactor design. The staff finds this acceptable.

3.3 Summary

Based on the above information, the NRC staff accepts FENOC's application to run the BVPS-1 with a power dependent PMTC. This power dependent PMTC is limited by the following: the MTC will not exceed $+0.2 \times 10^{-4}\ \Delta k/k/°F$ for all power levels up to 70% of RTP and the MTC limit will ramp linearly from $+0.2 \times 10^{-4}\ \Delta k/k/°F$ at 70% RTP to $0 \times 10^{-4}\ \Delta k/k/°F$ at 100% RTP.

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 (67 FR 58644). 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.

Principal Contributor: V. Klien

Date: February 6, 2003