August 2, 2005

Mr. L. William Pearce
Vice President
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Post Office Box 4
Shippingport, PA 15077

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2 (BVPS-1 AND 2) - REQUEST FOR ADDITIONAL INFORMATION (RAI) - EXTENDED POWER UPRATE (TAC NOS. MC4645 AND MC4646)

Dear Mr. Pearce:

By letter to the U.S. Nuclear Regulatory Commission (NRC) dated October 4, 2004, as supplemented February 23, May 26, June 14, and July 8, 2005, Agencywide Documents Access and Management System Accession Nos. ML042920300, ML051530376, ML051670270, and ML051940575, FirstEnergy Nuclear Operating Company (the licensee) submitted a license amendment request for BVPS-1 and 2 to change the operating licenses to increase the maximum authorized power level from 2689 megawatts thermal (MWt) to 2900 MWt which represents an increase of approximately 8 percent above the current maximum authorized power level. The NRC staff has determined that the additional information contained in the enclosure to this letter is needed to complete its review. As discussed with your staff, we request your response within 30 days of receipt of this letter, in order for the NRC staff to complete its scheduled review of your submittal.

If you have any questions, please contact me at 301-415-1402.

Sincerely,

/RA/

Timothy G. Colburn, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-334 and 50-412

Enclosure: RAI

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION

RELATED TO FIRSTENERGY NUCLEAR OPERATING COMPANY (FENOC)

BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2 (BVPS-1 AND 2)

EXTENDED POWER UPRATE (EPU)

DOCKET NOS. 50-334 AND 50-412

By letter dated October 4, 2004, as supplemented February 23, May 26, June 14, and July 8, 2005, Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML042920300, ML051530376, ML051670270, and ML051940575, FENOC (the licensee) proposed changes to the BVPS-1 and 2 operating licenses to increase the maximum authorized power level from 2689 to 2900 megawatts thermal rated thermal power or approximately 8 percent. The Nuclear Regulatory Commission (NRC) staff has reviewed the licensee's application against the guidelines in the EPU review standard (RS-001) and determined that it will need the additional information identified below to complete its review.

1. Section 10.16.1.2 of the risk assessment (Reference 2), states: "A review of the engineering change packages associated with the EPU including containment conversion was performed to determine their affect on systems and associated equipment that are important to plant risk."

    a. Are the BVPS-1 auxiliary feedwater cavitating venturis and main feedwater (MFW) fast-acting isolation valves related to EPU?

    b. For EPU-related change packages, please provide the details of these reviews for BVPS-1 and 2, including the effect of each modification on the probability risk assessment (PRA) model.

2. Section 10.16.1.4 of Reference 2, discusses the impact of EPU conversion on the human reliability analysis (HRA). The major impact is that the time available to perform some operator actions had decreased. In some cases, the base PRA model used a conservative estimate of the time available, which is taken in the analysis to bound the post-EPU time. The NRC staff notes that use of bounding times can mask the actual change in risk, although such practice should result in a bounding estimate of risk. The following clarifications and additional information are needed to facilitate determining the overall impact of EPU on the HRA.

    a. For both units, please provide the detailed HRA for all human interactions ("operator actions") that (1) have a Fussell-Vesely importance measure greater than 0.005 or a risk-achievement worth greater than 2, or (2) were modified to represent the post-EPU plant. Include whether the time available is considered "bounding" or is best estimate for pre- and post-EPU conditions.

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b. Table 10.16-5 provides post-EPU importance measures for selected operator actions. (1) Which unit PRA model was used to generate these importance measures? (2) Are the operator actions in this table, which are of the form "OPR*," the same as the corresponding actions in Table 10.16-2, which are designated "ZHE*" (where "*" represents an alphanumeric string).

c. Table 10.16-1 gives pre- and post-EPU times to core damage for station blackout scenarios. Why does this time increase on BVPS-1 and decrease on BVPS-2 for the "182 gpm, successful cooldown/depressurization, primary plant demineralized water storage tank make-up available" case?

d. Under the discussion of "general transients," it states: "Thus, with the RSG [replacement steam generators] there is less margin for successful completion of the plant-specific feed and bleed procedure ... initiated at 0.495 hours ...." Does the time available for this action change under EPU conditions? What is the human error probability (HEP) for this action, both pre- and post-EPU? Why was this action not included in Table 10.16-2 or 10.16-5?

e. Note 2 of Table 10.16-2 explains that the reduction in time available for a number of the operator actions is due to adopting a new reactor coolant pump seal loss-of-coolant accident model. Is this considered an EPU change?

f. Note 3 of Table 10.16-2 refers to changes in HRA because the pre-EPU model did not credit resetting containment isolation phase B. Is this considered an EPU change?

g. Note 4 of Table 10.16-2 says that ZHEIA1 is considered a "guaranteed success since the diesel air compressor will auto-start." Is this change due to a change to the plant equipment? Is it related to the EPU?

h. Table 10.16-5 shows the Fussell-Vesely importance of operator action OPRIA1, "Given LOSP [loss of offsite power], operators locally start the diesel air compressor," as 6.13E-04. Is this the same operator action as ZHEIA1 in Table 10.16-2? (It has the same description.) If "yes," how was the Fussell-Vesely determined, given that the HEP for ZHEIA1 is given as 0.0?

I. Section 10.15 of Reference 1 states: "A review of operating procedures/ emergency operating procedures/training potentially impacted by EPU will be completed ...." How was the full impact of the EPU on the human reliability analysis determined if operating procedure changes have not yet been identified?

j. Are there any additional operator actions that are considered in the model for estimating large early release frequency (LERF)? Please provide a listing of any operator actions unique to LERF and an assessment of the impact of the EPU on the corresponding HEPs.

3. Please provide an assessment of the increase in risk if only the EPU is considered. For example, the impact of containment conversion, BVPS-1 replacement steam

generators, BVPS-1 AFW cavitating venturis and MFW fast-acting isolation valves should not be included unless they are required for the EPU. Note that this can be done either by having non-EPU changes in both the base model and the post-EPU model or in neither.

The NRC staff would prefer that this assessment use realistic HEPs for both the pre-EPU and post-EPU analysis (where these would change) to avoid masking of the actual change in risk; refer to question 2, above. However, if bounding HEP numbers are employed, justify that the final risk metric is bounding with respect to those HEPs.

The following risk metrics should be provided for both BVPS-1 and 2:

a. Internal events core damage frequency (CDF) and LERF.

b. CDF and LERF from internal fires.

4. Section 10.16.1.5 states that the RSGs will result in a lower frequency for steam generator tube rupture (SGTR) because of the use of Alloy 690. Please provide the basis for the new SGTR frequency, including the supporting reference(s) (or excerpts).

5. What is the expected impact of EPU on the probability of a consequential loss of offsite power (LOOP)? For each unit, provide the contribution to the total CDF from consequential LOOP events in the current model. Provide the same information for operation at EPU conditions, or provide a sensitivity analysis showing how CDF would change assuming the probability of consequential LOOP increases after EPU.

6. The PRA results in the EPU risk assessment (Reference 2) were compared with those provided in a response to the NRC staff's questions on a recent license amendment request for extending the emergency diesel generator (EDG) allowed outage time (AOT) (Reference 3). The table below compares the information.

	EDG AOT (Ref. 3)	EPU (Ref. 2)
Beaver Valley Unit 1		
PRA Model Designator	BV1REV3	BV1REV3
Date Updated	9/2003	9/2003
CDF (per year)	2.34E-5	7.45E-6
LERF (per year)	1.03E-6	1.03E-6
Beaver Valley Unit 2		
PRA Model Designator	BV2REV3B	BV2REV3D
Date Updated	5/2003	5/2003
CDF (per year)	3.27E-5	2.01E-5
LERF (per year)	1.12E-6	1.12E-6

a. What has changed in the BVPS-1 and BVPS-2 PRA models since the Reference 3 letter?

b. Explain why BVPS-1 CDF has dropped significantly and BVPS-2 CDF has dropped somewhat compared to the Reference 3 values.

REFERENCES:

1. Letter from L. William Pearce, FirstEnergy Nuclear Operating Company, to U.S. Nuclear Regulatory Commission, “Beaver Valley Power Station, Unit No. 1 and No. 2 BV-1 Docket No. 50-334, License No. DPR-66 BV-2 Docket No. 50-412, License No. NPF-73 License Amendment Request Nos. 302 and 173,” L-04-125, October 4, 2004. (ADAMS Accession No. ML042920300)

2. Letter from L. William Pearce, FirstEnergy Nuclear Operating Company, to U.S. Nuclear Regulatory Commission, “Beaver Valley Power Station, Unit Nos. 1 and 2 BV-1 Docket No. 50-334, License No. DPR-66 BV-2 Docket No. 50-412, License No. NPF-73 Probabilistic Safety Review for License Amendment Request Nos. 302 and 173,” L-05-104, June 14, 2005. (ADAMS Accession No. ML051670270)

3. Letter from L. William Pearce, FirstEnergy Nuclear Operating Company, to U.S. Nuclear Regulatory Commission, "Beaver Valley Power Station, Unit No. 1 and No. 2 BV-1 Docket No. 50-334, License No. DPR-66 BV-2 Docket No. 50-412, License No. NPF-73 Response to Request for Additional Information in Support of LAR Nos. 306 and 176 Emergency Diesel Generator Allowed Outage Time Extension," L-04-141, October 29, 2004. (ADAMS Accession No. ML043070444)

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