November 1, 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 Nuclear Regulatory Commission (NRC) dated October 4, 2004, as supplemented February 23, May 26, June 14, July 8, September 6, and October 7, 2005, Agencywide Documents Access and Management System, Accession Nos. ML051160426, ML042920300, ML051530376, ML051670270, ML051940575, ML052550373, and ML052850145, 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
Plant Licensing Branch A
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-334 and 50-412

Enclosure: RAI

cc w/encl: See next page

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SECOND-ROUND REQUEST FOR ADDITIONAL INFORMATION (RAI)

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, July 8, September 6, and October 7, 2005, Agencywide Documents Access and Management System, Accession Nos. ML051160426, ML042920300, ML051530376, ML051670270, ML051940575, ML052550373, and ML052850145, FENOC (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 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. These questions reference licensee's responses from the licensee's May 26, and July 8, 2005, RAI responses.

May 26, 2005, Second-Round RAI Questions

1. In response to the NRC staff's RAI Question No. N1, you indicated that the main steamline breaks and feedwater line breaks were not considered in the EPU analysis because the loss-of-coolant accident (LOCA) loads due to these breaks have no significant impact on the reactor vessel (RV) and internals. Confirm whether these LOCA loads at the EPU conditions are bounded by the original design-basis forcing functions. If not, justify not applying these loads to the dynamic model and evaluate the impact on the stresses and cumulative usage factor (CUF) calculations, especially for the steam generator internals, shell and supports.

2. In your response to RAI Question No. N4, you described the dynamic model, method, and analysis in support of the EPU and replacement steam generators (RSGs) at BVPS-1. You also indicated that the results from the NUPIPE-SWPC analyses for the primary reactor coolant loop piping include loads on the major components, nozzles, and supports for the normal operating conditions, upset conditions, and the faulted LOCA conditions. These loads were used in the evaluations of the major components, nozzles, and supports (including the RSGs, reactor coolant pumps, and RV), and have been shown to be acceptable. You also indicated that the stresses and CUFs provided for BVPS-1 piping, components and supports, including the RV and internals, at the EPU conditions include the dynamic effect of the RSGs. However, in response to RAI Question No. N5, you indicated that:

   > "...the evaluation provided in Section 4.7.1 of the EPU Licensing Report is qualitative since, as stated in the introduction to this section, the licensing

Enclosure

> acceptability of replacing the Model 51 original SG components with Model 54F replacement SG components is being evaluated under the provisions of 10 CFR 50.59 [Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 59]. ... The stress information requested is being generated as part of the design process for the BV-1 replacement steam generators. The information will be included in the Design Stress report. When completed, it will be available on-site for NRC review and inspection."

Please explain the apparent discrepancy. The information that was requested in Question No. N5 for BVPS-1 is needed for the NRC staff to complete its evaluation. The information requested is similar to that provided to support the BVPS-2 original SGs operating at the EPU conditions.

3. In the first paragraph of its response to Question No. N9 in Enclosure 1 to its letter dated May 26, 2005, FENOC states that the worst-case scenarios that determine maximum differential pressure across each motor-operated valve (MOV) are unaffected by the EPU. In the second paragraph of that response, FENOC states that the EPU requires a change to the high head safety injection (HHSI) pump and that this change potentially increases the differential pressure across various valves. Discuss these two responses.

4. In Item B of its response to Question No. N10, FENOC refers to its application of the ComEd pressure-locking thrust prediction methodology for MOVs-1SI-869A/B. Discuss the application of this method.

5. In the first bullet of Item C of its response to Question No. N10, FENOC states that the safety injection (SI) system valves were modified to eliminate the potential for by drilling a hole in one disc of each valve. Discuss how this modification eliminates the potential for thermal binding.

6. In Item 2 of its response to Question No. N11 on air-operated valves (AOVs), FENOC states that the feedwater regulating valves (FWRVs) FCV-1FW-478, 488, and 498, and 2FWS-FCV-478, 488, and 498 have increased flow requirements under EPU conditions. FENOC states that the Unit 1 FWRVs were modified and the Unit 2 FWRVs are being replaced. Discuss the qualification of these valves to perform their safety functions.

7. In the first bullet in its response to Question No. N12 on the Inservice Testing (IST) Program, FENOC states that new fast-acting Feedwater Isolation Valves HYV-1FW-100A, B, and C have been installed. Discuss the qualification of the capability of these valves to perform their safety functions.

8. In the fourth bullet in its response to Question No. N12, FENOC states that the tolerance settings for the main steam safety valves (MSSVs) and pressurizer safety valves were increased for the EPU. FENOC states MSSVs with the lowest setting pressure will be limited to a lift-setting tolerance of +1/-3%. The lift-setting tolerance for the remaining MSSVs will be limited to a lift-setting tolerance of ±3%, which is a change from the current lift-setting tolerance of +1/-3%. The upper tolerance for the pressurizer code safety valves will be changed from +1% to +3% for BVPS-1, and from +1% to +1.6% for BVPS-2. The current lift-setting tolerance for the pressurizer code safety valves for both units is +1%/-3%. The lower tolerance for the pressurizer code safety valves for both units is unchanged at -3%. Discuss the impact of these changes on safety margins.

9. Is FENOC relying on safety valves at BVPS-1 and 2 to operate with water flow for EPU conditions? If so, discuss the qualification of the valves for this condition.

10. In its response to Question No. N13 on monitoring potential adverse flow effects during EPU startup, FENOC provides examples of its Level 2 Acceptance Limits. In the fifth bullet of those examples, the licensee states that visual observations will be made of increased pipe or component vibration. Discuss the adequacy of visual observations in lieu of the use of accelerometers.

11. In its response to Question No. N14, FENOC discusses its consideration of potential flow-induced vibration effects. Has FENOC addressed the capability of any feedwater or condensate sample probes to withstand increased flow under EPU conditions?

July 8, 2005, Second-Round RAI Questions

1. Overpressure Protection During Power Operation

   The BVPS-1 and 2 EPU submittal does not address the analysis requirements of Standard Review Plan (SRP), Section 5.2.2, Section II.A, “Overpressure Protection.”

   The BVPS-1 EPU submittal, and L-05-112 Enclosure 2 (Response B.1) deals with overpressure protection during power operation (i.e., safety valve sizing, by analyzing a Chapter 15 event). Chapter 15 event analyses do not address the overpressure protection requirements of the American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code* (ASME Code), Section III, NB-7300 and NC-7300, and they are not consistent with the approach described in WCAP-7769, Rev 1, “Overpressure Protection for Westinghouse Pressurized Water Reactors,” June 1972.

   Please provide analyses, per SRP 5.2.2, II.A, which show the continued adequacy of the BVPS-1 safety valve sizing, and consistency with the BVPS-1 licensing basis-design analysis approach of WCAP-7769, Rev 1, which credits the second safety-grade trip from the reactor protection system.

   The same question applies to BVPS-2. Please provide analyses, per SRP 5.2.2, II.A, which show the continued adequacy of the BVPS-2 safety valve sizing, and consistency with the BVPS-2 licensing basis-design analysis approach of WCAP-7769, Rev 1, which credits the second safety-grade trip from the reactor protection system.

2. If an inadvertent emergency core cooling system (ECCS) actuation event should occur, and the block valves are open, then the power-operated relief valves (PORVs) might open and possibly relieve water. If the PORVs are assumed to reseat properly, after having relieved water, please provide information that details how the circuitry that controls the closing signal will meet Class 1E requirements. This is necessary in order to justify the assumption that the PORVs can be relied upon to close when pressurizer pressure drops below the closing setpoint.

3. For the analysis case in which the block valves are assumed to be closed (i.e., the case that assumes the PORVs are not credited to mitigate an inadvertent ECCS actuation event), please explain how the pressurizer water temperature is calculated. Specifically, indicate whether the water in the pressurizer is assumed to be uniformly mixed with the

insurge water, or stratified (i.e., simply pushed out of the safety valves in a piston fashion). Also please discuss the basis for assuming pressurizer heaters operate when the reactor coolant system (RCS) is pressurizing.

4. Please provide copies of the following documents, all of which are cited in Enclosure 1, Attachment C:

   a. OE8903 (Potential for RCS to be outside the Design Basis during an Inadvertent ECC Actuation at Power) - Diablo Canyon Issue

   b. CR 980894 (Evaluation of Diablo Canyon pressurizer safety (PSV) valve issue for BVPS)

   c. NSAL 98-007, 8/11/98 - PSV Evaluations with modified pressurizer heater and spray models

   d. EM 116856 (Evaluation of NSAL 98-007 for BVPS)

   e. Letter NPDDBE;0069 - 5/11/98 - R. A. Hruby to K. L. Ostrowski (PSV issue not applicable to BVPS-1).

   f. Westinghouse calculations CN-TA-98-031 and 032 which provide results of PSV operability studies for inadvertent ECCS and feedline break events (limiting events for pressurizer fill/PSV water relief).

   g. Westinghouse letter DLC-98-736 (N.S. Kury to W.R. Kline, 6/4/98) which transmitted the Westinghouse calculations and confirmed acceptability of the calculation results

5. FENOC's Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.59, Evaluation No. 98-258 (L-05-112 Enclosure 1, Attachment C), applies to BVPS-2. Please provide either a corresponding evaluation for BVPS-1 or an explanation of why such an evaluation is not applicable (i.e., based upon Letter NPDDBE;0069 - 5/11/98).

6. Section 5.3.18 of the license amendment request (LAR) states, "The third criterion is met if it can be demonstrated that the pressurizer does not become water-solid in the minimum allowable operator action time. However, if SI flow is not terminated before the pressurizer becomes water solid, it must be demonstrated that this Condition II event does not lead to a more serious plant condition. In this situation, a pressurizer safety valve (PSV) operability analysis must be performed to demonstrate that the PSVs would continue to function under water relief conditions for the period of time required for the operators to take action to prevent or terminate water relief through the safety valves. The required operator action is to either terminate SI flow to avert a water-solid condition or to confirm that at least one PORV is unblocked and available for relief. Should water relief through the pressurizer PORVs occur, the PORV block valves would be available to isolate the RCS if a PORV fails to close."

   a. What are the procedures that direct the operator to, "either terminate SI flow to avert a water-solid condition or to confirm that at least one PORV is unblocked

and available for relief"? How long would it take the operator, following procedures, to accomplish these actions?

b. If the PSVs are qualified for water relief, then why would the operator have to, "either terminate SI flow to avert a water-solid condition or confirm that at least one PORV is unblocked and available for relief"? If the PSVs are qualified for water relief, then it seems the operator would simply be required to terminate charging flow before pressurizer water temperature drops below the temperature shown to be acceptable in the Electric Power Research Institute's (EPRI's) valve tests. Where is the time limit and procedure for that?

c. How does the BVPS-1 and 2 application, and subsequent RAI responses, demonstrate that the inadvertent ECCS actuation, a Condition II event, does not lead to a more serious Condition III event?

7. BVPS-1 and 2 current Technical Specification (CTS) 3.9.8.1, "Residual Heat Removal and Coolant Circulation," Action C, states, "The residual heat removal loop may be removed from operation for up to 4 hours per 8 hour period during the performance of Ultrasonic In-service Inspection inside the reactor vessel nozzles provided there is at least 23 feet of water above the top of the reactor vessel flange." This residual heat removal (RHR) out of service allowance is four times longer than the standard technical specification (STS) allowance. In order for the NRC staff to continue their review of FENOC's EPU request for BVPS-1 and 2, provide the analysis that demonstrates that the 4-hour RHR out-of-service allowance is acceptable. Include inputs, assumptions, methodologies, and limitations on the analysis. In addition, please identify any and all limitations and restrictions on the use of the 4-hour RHR out-of-service allowance.

   Additionally, CTS 3.9.8.1 Action C, BVPS-1 and 2 CTS 3.9.8.1, "Residual Heat Removal and Coolant Circulation," includes Action B which states, "The residual heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel (hot) legs." In order for the NRC staff to continue their review of FENOC's EPU request for BVPS, provide the following:

   a. Describe the BVPS-1 and 2 controls that prevent the 3.9.8.1 Action B exception that allows the required RHR loop to be removed from operation for #1 hour per 8-hour period, and the 3.9.8.1 Action C exception that allows the required RHR loop to be removed from operation for#4 hours per 8-hour period from being invoked during the same 8-hour period and/or consecutively.

   b. The analysis which shows the RHR requirements continue to be met with the worst case synergistic effects of these exceptions. Include all inputs, assumptions, limitations, and results of that analysis. Identify any controls necessary to ensure the analysis remains bounding.

8. The response to Question No. A13, dated July 8, 2005, states, "The current spent fuel pool criticality licensing basis for BVPS-1 and 2 does not include a commitment to 10 CFR 50.68." It continues to imply that both BVPS-1 and BVPS-2 are operating under a 10 CFR 70.24 exemption. BVPS-2 takes credit for the presence of soluble boron, which is allowed only under 10 CFR 50.68, as stated in the safety evaluation for

Amendment No. 128, dated February 11, 2002 (ADAMS Accession No. ML0200203731).

Since the 10 CFR 70.24 exemption was issued prior the allowance for boron credit, it is no longer valid. It is necessary to commit to 10 CFR 50.68. Please demonstrate that the requirements of 10 CFR 50.68 are satisfied, or document that the 10 CFR 70.24 exemption allows for boron credit.

9. This question refers to the licensee's response to Section 5.4, question X.1, of Enclosure 2 of the July 8, 2005, RAI response (pages 294-295). In the EPU report for BVPS-1 and 2, the steam generator tube rupture (SGTR) analysis is based on the assumption that the leak flow from the RCS to the secondary side of the SG is terminated 30 minutes following the event initiation. In response to the NRC staff questions regarding the adequacy of the assumed 30-minute time for terminating the break flow, it is indicated that the Updated Final Safety Analysis Report (UFSAR) was changed to reflect a 51-minute termination time via 10 CFR 50.59. However, the licensee stated that the use of a 30-minute termination time assumed in the methodology still results in a more conservative analysis with respect to the offsite dose consequence analysis. Please prove clarification to substantiate this conclusion.

   It is stated in the July 8, 2005, RAI response that a supplemental SGTR analysis has been performed for BVPS-1 that includes the most limiting single failure, coincident with a loss-of-offsite power (LOOP), and with operator actions as assumed in the emergency operating procedures. This supplemental analysis confirmed the conservatism of dose calculations based on the 30-minute termination-of-event assumption. It is also stated that supplemental SGTR analyses have been performed to demonstrate margin to SG overfill for BVPS-1 with various single-failure assumptions considered. Please provide the results of these supplemental SGTR analyses including major assumptions, analyses methodology used, and transient curves developed for the NRC staff to review.

cc:

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