**FENOC**

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L-01-078

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

**Subject: 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 a Request for Additional Information**
**In Support of LAR Nos. 289 and 161**

This letter provides the FirstEnergy Nuclear Operating Company (FENOC) response to a NRC Request for Additional Information (RAI) in support of License Amendment Requests (LAR) 289 and 161. The LARs were submitted by FENOC letter L-01-006 dated January 18, 2001 and propose a 1.4% power uprate for both Beaver Valley Power Station (BVPS) units.

The RAI solicits details regarding the following at the uprated power level:

- relief capacity for all main steam safety valves,
- design bases transient and accident evaluations,
- adequacy of the safety related condensate storage tank volume, and
- effects on ATWS analyses.

The FENOC responses are provided in Attachment A of this letter. FENOC requests NRC approval of License Amendment Requests 289 and 161 to support implementation of the power uprate for the summer of 2001. An implementation period of up to 60 days is requested following the effective date of this amendment.

This information does not change the evaluations or conclusions presented in FENOC letter L-01-006. If there are any questions concerning this matter, please contact Mr. Thomas S. Cosgrove, Manager Regulatory Affairs at 724-682-5203.

Sincerely,

Lew W. Myers

Lew W. Myers

Attachment

A001

c: Mr. L. J. Burkhart, Project Manager
Mr. D. M. Kern, Sr. Resident Inspector
Mr. H. J. Miller, NRC Region I Administrator
Mr. D. A. Allard, Director BRP/DEP
Mr. L. E. Ryan (BRP/DEP)

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I, Lew W. Myers, being duly sworn, state that I am Senior Vice President of FirstEnergy Nuclear Operating Company (FENOC), that I am authorized to sign and file this submittal with the Nuclear Regulatory Commission on behalf of FENOC, and that the statements made and the matters set forth herein pertaining to FENOC are true and correct to the best of my knowledge and belief.

FirstEnergy Nuclear Operating Company

Lew W. Myers
Lew W. Myers
Senior Vice President - FENOC

COMMONWEALTH OF PENNSYLVANIA

COUNTY OF BEAVER

Subscribed and sworn to me, a Notary Public, in and for the County and State above named, this 9 th day of June, 2001.

Sheila M Fattore
My Commission Expires:

Notarial Seal
Sheila M. Fattore, Notary Public
Shippingport Boro, Beaver County
My Commission Expires Sept. 30, 2002
Member, Pennsylvania Association of Notaries

## NRC Request for Additional Information

### NRC Request Number 1

The proposed technical specification (TS) bases B 3/4.4.7.1.1 indicates that the total relieving capacity for all main steam safety valves (MSSVs) is 108% of the total steam flow at rated thermal power. This capacity has been reduced from the current value of 110%. Provide justification of this proposed change in light of ASME code requirements for safety valves.

### FENOC Response to Request Number 1

Five Main Steam Safety Valves (MSSVs) are located on each main steam header outside containment, upstream of the main steam isolation valves. The valve lift settings and relieving capacities are specified in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition and Winter 1972 Addenda.

The ASME B&PV Code Section III does not require 110% relieving capacity. Rather, the Code requires that sufficient capacity exists to prevent pressure from exceeding 110% of design pressure during design transients. The Nuclear Steam Supply System (NSSS) secondary system is designed for 1,100 psia and 560°F. This design basis is sufficient to cope with any anticipated operational occurrence or Design Basis Accident (DBA).

Transient analyses are performed to demonstrate that the Code requirement is met. The worst case event for main steam system pressure is the Loss of Load event (with accompanying loss of condenser vacuum and consequently loss of steam dump capability). Analysis of this event at the uprated power level shows that the peak main steam pressure remains below 1208.5 psia, and therefore less than 110% of system design pressure. This satisfies the criteria in the relevant ASME piping codes, B31.1 and ASME B&PV Code Section III (See Table A-1).

### NRC Request Number 2

It is indicated in your submittal that the design bases transients and accidents have been evaluated at the uprated power level and the results of the analyses demonstrated that all the applicable acceptance criteria for each event continued to be met at the 1.4% power uprate conditions (considering the updated primary and secondary system temperatures, pressures, flows, etc). Please provide detailed results of the re-analyses in the following areas:

a) Major assumptions used in the re-analyses. Provide justification for any assumptions which are deviate from that used in the existing analyses.

b) Describe methods and computer codes used for the re-analyses and confirm that they are previously approved by the staff. Provide justification for any changes in methodology from the existing analyses.

c) Provide the results of the re-analyses including primary and secondary system peak pressure, minimum DNBR, and/or amount of failed fuel.

**FENOC Response to Request Number 2a**

Major assumptions made for the re-analyses are identical to those made in the existing analyses with the exception of assumptions covered by the Revised Thermal Design Procedure (RTDP) methodology. The RTDP methodology only affects Departure from Nucleate Boiling (DNB) events initiated at power. In the RTDP analyses, uncertainties on initial conditions are included in the safety analysis limit Departure from Nucleate Boiling Ratio (DNBR) rather than being explicitly included in the transient initial conditions. This is discussed in detail in the RTDP topical report WCAP-11397-P-A and WCAP-11397-A (April 1989) and letter, A. C. Thadani (USNRC) to W. J. Johnson (Westinghouse), "Acceptance for Referencing of Licensing Topical Report WCAP-11397, Revised Thermal Design Procedure," (January 1989).

**FENOC Response to Request Number 2b**

With respect to the non-LOCA safety analyses, the only revised methodology is implementation of the Revised Thermal Design Procedure (RTDP), which is described in detail in WCAP-11397-P-A and WCAP-11397-A (April 1989). The computer codes used for the non-LOCA transient analyses are the same codes that were used for the existing analyses. Specifically, these codes are LOFTRAN (WCAP-7907-P-A and WCAP-7907-A, April 1984), FACTRAN (WCAP-7908-A, December 1989) and TWINKLE (WCAP-7979-P-A, November 1972 for Beaver Valley Power Station (BVPS) Unit 1, and WCAP-8028-A, January 1975 for BVPS Unit 2).

**FENOC Response to Request Number 2c**

For the 1.4% uprating and RTDP, only DNB events needed to be reanalyzed. The existing analyses for the non-DNB events, including LOCA, remain applicable. The existing analyses of record are initiated at 2713 MWt, 102% of the pre-uprated nominal power, which is equivalent to 100.6% of the uprated NSSS power.

With the Revised Thermal Design Procedure, initial condition uncertainties are statistically combined in the safety analysis limit DNBR. As such, the nominal power level is assumed in the analyses for any event covered by RTDP methods (i.e., DNB events) and, thus, only the DNB events needed to be reanalyzed in support of RTDP and 1.4% uprating. Plant specific margin to accommodate rod bow and other DNB penalties and allow for flexibility in the design, operation and analysis of the plant is provided by performing the safety analyses to a DNBR limit value of 1.36. The Updated Final Safety Analysis Report (UFSAR) sections (UFSAR section 3.4.1.1 for BVPS Unit 1 and UFSAR section 4.4.1.1 for BVPS Unit 2) will be revised to reflect the new DNBR design and safety analysis limits associated with the RTDP methodology.

Table A-1 summarizes the results of the non-LOCA analyses performed in support of the RTDP License Amendment Request (LAR).

**NRC Request Number 3**

In Section 3.7.4 of Enclosure 1 of your submittal, discuss the affect from higher decay heat to the adequacy of the safety related condensate storage tank volume in light of:

a) to support AFW for achieving plant cooldown to RHR initiation, and

b) to assure SBO coping analysis remain valid.

**FENOC Response to Request Number 3**

a) The required inventory for achieving plant cooldown to Residual Heat Removal (RHR) initiation temperature is based on removal of decay heat, reactor coolant pump heat and sensible heat from the Reactor Coolant System (RCS). The current calculation of record considers operation at 102 percent of the engineered safeguards design rated power and therefore the results are not affected by the 1.4% uprate. This inventory requirement is less limiting than the current Technical Specification basis requirement for 9 hours at hot standby.

b) The required inventory to assure the Station Blackout (SBO) coping analysis remains valid is based on adequate condensate for a 4 hour station blackout duration. The higher decay heat as a result of the 1.4% uprate will increase this required inventory. This increase was evaluated and the results showed that required inventory is still within the minimum Technical Specification limits for condensate inventory.

**NRC Request Number 4**

Please submit information that discuss effect of power uprate on ATWS analyses, including any changes in important core or energy release assumptions.

**FENOC Response to Request Number 4**

The current licensing basis safety analyses for BVPS Units 1 and 2 support operation at a nominal reactor power of 2652 MWt (2660 MWt NSSS power) and includes consideration of a maximum 2% uncertainty on power. Hence, the current BVPS licensing basis supports operation at a maximum reactor power level of 2705 MWt with uncertainty on power included.

For Anticipated Transients Without Scram (ATWS), operation of the BVPS units at the current licensed power level is supported by the Westinghouse generic ATWS analyses performed in accordance with the guidelines provided in NUREG-0460. These analyses, which are documented to the NRC in Westinghouse letter NS-TMA-2182 dated December 31, 1979, also are the analytical basis for the Final ATWS Rule, 10CFR50.62(b), as applicable to Westinghouse Pressurized Water Reactors (PWRs). Consistent with the guidelines prescribed in NUREG-0460, these ATWS analyses were performed assuming initial operating conditions consistent with nominal plant conditions. In the analyses presented in NS-TMA-2182, a nominal reactor power of 2785 MWt was assumed for the generic 3-Loop Westinghouse PWR configuration applicable to the BVPS units. As prescribed by NUREG-0460, the ATWS analyses presented in NS-TMA-2182 also included sensitivity analyses for variations in specific parameters. Included in these sensitivity analyses are the results of varying reactor power by + 2%. Hence, the results of the ATWS analyses presented in NS-TMA-2182 support operation at a maximum reactor power of 2841 MWt for the 3-Loop PWR plant configuration.

For the subject BVPS Unit 1 and 2 power uprate, the maximum reactor power level, including uncertainty, remains unchanged. The 1.4% power uprate is achieved by reducing the uncertainty on power from 2% to 0.6% and maintaining the same net maximum reactor power with uncertainty. With the 1.4% power uprate, the nominal reactor power level increases from 2652 MWt to 2689 MWt. This nominal power level at the uprate condition remains bounded by the nominal reactor power of 2785 MWt assumed in the generic ATWS analyses. Hence, in response to this request, the applicable information relative to ATWS is that provided in NS-TMA-2182.

**Table A-1**
**Summary of the non-LOCA analyses performed in support of the RTDP Methodology.**

| Event Name | UFSAR Section | Minimum DNBR | Peak Primary Pressure | Peak Secondary Pressure |
|---|---|---|---|---|
| Rod Withdrawal at Power (Unit 1) | 14.1.2 | 1.370 | N/A * | 1171 psia |
| Rod Withdrawal at Power (Unit 2) | 15.4.2 | 1.362 | N/A * | 1171 psia |
| Partial Loss of Flow (Unit 1) | 14.1.5 | 1.787 | 2339.5 psia | 922.2 psia |
| Partial Loss of Flow (Unit 2) | 15.3.1 | 1.790 | 2327.8 psia | 920.6 psia |
| Loss of Load - DNB Case (Unit 1) | 14.1.7 | 1.72 | 2647.1 psia | 1177.2 psia |
| Loss of Load - DNB Case (Unit 2) | 15.2.2/15.2.3 | 1.67 | 2747.5 psia | 1182.5 psia |
| Feedwater Malfunction (Unit 1) | 14.1.9 | 1.835 | 2338 psia | 1123 psia |
| Feedwater Malfunction (Unit 2) | 15.1.1/15.1.2 | 1.894 | 2341 psia | 1179 psia |
| Excessive Load Increase (Unit 1) | 14.1.10 | >1.36 | N/A | N/A |
| Excessive Load Increase (Unit 2) | 15.1.3 | >1.36 | N/A | N/A |
| RCS Depressurization (Unit 1) | 14.1.15 | 1.65 | N/A | N/A |
| RCS Depressurization (Unit 2) | 15.6.1 | 1.76 | N/A | N/A |
| Complete Loss of Flow (Unit 1) | 14.2.9 | 1.36 | 2421.1 psia | 949.4 psia |
| Complete Loss of Flow (Unit 2) | 15.3.2 | 1.36 | 2141.2 psia | 951.0 psia |
| Limits | --- | 1.36 | 2748.5 psia | 1208.5 psia |

| Event Name | UFSAR Section | Percentage of rods in DNB | Peak Primary Pressure |
|---|---|---|---|
| Locked Rotor - DNB Case (Unit 1) | 14.2.7 | < 18% | 2691 psia |
| Locked Rotor - DNB Case (Unit 2) | 15.3.3 | < 18% | 2759.3 psia |
| Limits | --- | 18% | 2997 psia** |

* A generic Westinghouse evaluation addresses peak pressures for Rod Withdrawal at Power analyses.

** The peak Reactor Coolant System pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits.