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L-01-108

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:

- original vessel outlet temperatures used for both Units 1 and 2,
- variation of the vessel inlet temperature,
- NSSS design parameters,
- scaling factors, and
- NSSS piping systems.

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

A001

Attachment

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 21 th day of August, 2001.

Vivian V. Harper

My Commission Expires:

Notarial Seal
Vivian V. Harper, Notary Public
Ohioville Boro, Beaver County
My Commission Expires July 28, 2003
Member, Pennsylvania Association of Notaries

Response to Request For Additional Information
Proposed Amendment for Power Uprate
Beaver Valley Power Station, Units 1 And 2

1. Table 3-1 of Enclosure 1 to the reference transmittal provides the NSSS design parameters that are used as the basis for the 1.4 percent power uprate for Beaver Valley Units 1 and 2. In Section 3.6.1, you stated that the vessel outlet temperature increases from 610.4°F to 610.8°F and the vessel inlet temperature decreases from the current 542.0°F to 541.6°F as a result of the 1.4 percent uprate program. Therefore, both the T_{hot} and T_{cold}, variation during normal plant loading and plant unloading are increased. You also stated that the vessel outlet temperature associated with the 1.4-percent power uprate is less than the vessel outlet temperature that was originally analyzed for the Unit I reactor vessel outlet nozzles. Was Unit 2 also using a higher vessel outlet temperature in the original design basis analyses than the vessel outlet temperature associated with the 1.4-percent power uprate in Table 3-1? Provide the original vessel outlet temperatures used for both Units 1 and 2. Also, confirm that there is no change in core flow rate, and LOCA loads.

   Response:

   The vessel outlet temperatures used for both Units 1 and 2 are discussed under question 2.

   There are no changes in the core flow rate associated with the 1.4% uprate.

   A review of the applicable LOCA hydraulic forcing functions with respect to the uprate conditions was conducted. The increase in T_{hot} has no negative impact on LOCA forces. The 0.4°F decrease in T_{cold} results in a very small increase in the magnitude of LOCA forces. The analyses of record were reviewed with respect to available margins with the following conclusions.

   The Beaver Valley Unit 1 vessel LOCA forces remain bounding with a conservatively estimated margin of 4.7% to the uprated conditions. Beaver Valley Unit 1 fuel qualification was performed using LOCA forces generated to be bounding for both Beaver Valley Unit 1 and other plants with similar vessel designs for a vessel inlet temperature of 522.6°F, which substantially bounds the temperatures for the Unit 1 uprate. The Beaver Valley Unit 1 loop and steam generator LOCA forces remain bounding for the uprated conditions with conservatively estimated margins of 4% and 14%, respectively.

   The Beaver Valley Unit 2 vessel and fuel LOCA forces remain bounding with a conservatively estimated margin of 6%. The Beaver Valley Unit 2 loop and steam generator LOCA forces remain bounding with conservatively estimated margins of 15% and 14%, respectively.

2. In Section 3.6.1, you indicated that at Unit 2, the vessel inlet temperature associated with the 1.4-percent power uprate provides a temperature variation of 5.4°F during plant loading and unloading. This magnitude of temperature change is less than the 7.0°F change in T_{cold} considered for plant loading and unloading in the original reactor vessel stress report. Therefore, the effects of the revised T_{cold} variation during plant loading and unloading are considered to be bounded by the original analysis. Confirm whether at

Unit 1, the variation of the vessel inlet temperature associated with the 1.4 percent power uprate is also bounded by the original analysis.

Response:

Unit 1

The Beaver Valley Unit 1 reactor vessel was originally analyzed by Combustion Engineering, Inc. (CE) in accordance with its reactor vessel equipment specification to operate with a normal operating inlet temperature of 543.5°F and a normal operating outlet temperature of 610.9°F. This original design analysis is documented in the CE analytical report and addenda. The normal operating temperatures were modified by the 20% Steam Generator Tube Plugging (SGTP) Program performed in 1991. The vessel inlet temperature decreased by 1.0°F to 542.5°F, and the vessel outlet temperature also decreased by 1.0°F to 609.9°F. These temperature changes were evaluated and justified in the reactor vessel evaluation for 20% SGTP. The vessel temperatures were further modified by reduced thermal design flow in conjunction with 20% SGTP in 1992. With this program, the vessel inlet temperature further decreased by 0.3°F to 542.2°F, and the vessel outlet temperature increased up to 610.6°F. The revised vessel inlet temperature was considered in the reactor vessel evaluation, but the vessel outlet temperature remained bounded by the previous analyses. The tube plugging level was increased to 30% with reduced thermal design flow in 1993, and the vessel inlet temperature was reduced by another 0.2°F to 542.0°F while the vessel outlet temperature remained bounded by the original 610.9°F. Therefore, the Beaver Valley Unit 1 reactor vessel was analyzed for vessel inlet temperatures from 542.0°F up to 547.0°F zero load temperature, and vessel outlet temperatures up to 610.9°F with the corresponding design transients prior to the 1.4% uprate. The 1.4% uprate program with 0% to 30% SGTP, reduced thermal design flow and RFA fuel with IFMs further reduced the vessel inlet temperature to 541.6°F while the vessel outlet temperature remained less than the analyzed value at 610.8°F. Therefore, the Beaver Valley Unit 1 reactor vessel was evaluated only for the 0.4°F reduction in vessel inlet temperature for the 1.4% uprate program.

Unit 2

The Beaver Valley Unit 2 reactor vessel was originally analyzed by Combustion Engineering, Inc. (CE) in accordance with its reactor vessel equipment specification to operate with a normal operating inlet temperature of 550.0°F, a normal operating outlet temperature of 620.0°F and a zero load temperature of 557.0°. This original design analysis is documented in the CE analytical report and addenda. The vessel inlet temperature of 542.5°F and the vessel outlet temperature of 609.9°F along with zero load temperature of 547°F were evaluated and justified the reactor vessel evaluation for the 20% Steam Generator Tube Plugging (SGTP) Program. This evaluation was performed by establishing 547.0°F as the zero load temperature and shifting the normal operating temperature range accordingly based upon the original range. Thus, the normal operating inlet temperature became 540.0°F and the normal operating vessel outlet temperature became 610.0°F. The vessel temperatures were further modified by reduced thermal design flow in conjunction with 20% and 30% SGTP in 1993. With this program, the PCWG vessel inlet temperature further decreased by 0.5°F to 542.0°F, and the vessel outlet temperature increased up to 610.6°F. The reactor vessel was evaluated for the revised vessel outlet temperature while the vessel inlet temperature remained bounded. Therefore, the Beaver Valley Unit 2 reactor vessel was analyzed for vessel inlet temperatures from 540.0°F up to the 547.0°F zero load temperature, and vessel outlet temperatures up to 610.6°F with the corresponding design transients prior to the 1.4% uprate. The 1.4% uprate program with 0% to 30% SGTP, reduced thermal design flow and RFA fuel with IFMs further reduced the vessel inlet temperature to 541.6°F while the vessel outlet temperature increased to 610.8°F. Therefore, the Beaver Valley Unit 2 reactor vessel was only evaluated for the 0.2°F increase in vessel outlet temperature for the 1.4% Uprate program.

3. In Section 3.6.3.3, you indicated that the primary input to the evaluations of the reactor internals are the NSSS design parameters given in Table 3-1 and the gamma heating rates. Provide a summary of evaluation results including the maximum calculated stresses and cumulative fatigue usage factors (CUFs) for the critical reactor internal components including the baffle/barrel region components, core barrel, baffle plate, baffle/former bolts, and lower core plate for the 1.4 percent uprated power conditions. Also provide the Code and Code Edition used for the evaluation of the reactor internal components. If different from the Code of record, please justify and reconcile the differences. Also, confirm that there is no increase in the potential for flow induced vibration.

<u>Response:</u>

The stresses and CUF for the reactor internal components were not adversely affected by the 1.4% uprate condition. There were no changes to design transients, vessel forces, and fuel characteristics from the 1.4% uprate program. The effect of gamma heat generation increases, from the 1.4% uprate, on the baffle/barrel region components, (core barrel, baffle plate, baffle/former bolts) was bounded by existing analyses. The increase in gamma heat generation seen by the lower core plate was evaluated. The increase to the total fatigue usage factor for the lower core plate due to the 1.4% power uprating was insignificant. For the lower core plate evaluation, the ASME Boiler and Pressure Vessel Code Section III Division I 1989 Edition was used.

The reactor internals are not licensed to a Code Edition. The reactor internal components were originally designed based on sound engineering practice.

The only change from the 1.4% uprate program that affects internals flow induced vibration is the increase to the fluid density. The current flow induced vibration evaluation bounds this change. Therefore, there is no effect to the current flow induced vibration evaluation due to the 1.4% uprate program.

4. In reference to Section 3.6.7, you stated that since certain operating parameters will change due to the 1.4 percent power uprate and 30 percent steam generator tube plugging, scale factors were developed based on the change in operating conditions. The scale factors were applied to the baseline analysis results to develop revised stresses and fatigue usage. Discuss the method, assumptions and technical basis regarding the calculation of the scaling factors, and provide the ASME Code Edition and Addenda used for the evaluation. If different from the Code of record, justify and reconcile the differences. Also, confirm that there is no increase in the potential for flow induced vibration of the steam generator U-bend tubes due to the proposed power uprate.

Response:

Calculation of scale factors:

Primary side components: The stress in primary side components is dependent on the differential pressure between the primary side and secondary side. The stresses are scaled from the generic Model 51 stress reports for calculating 1.4 % uprate condition.

Case 1: Stress Report Case: P_{pri} = 2250 psi, P_{sec} = 790 psi (100 % power)

Primary pressures, T_{stm} , P_{stm} are calculated for all the transients based on 100 % power condition for stress report case and the differential pressure between the primary and secondary side is calculated for all the transients.

Case 2 : Uprate 1.4 %, 30 % plugging: P_{pri} = 2250 psi, P_{sec} = 716 psi (100 % power)

Primary pressures, T_{stm}, P_{stm} are calculated for all the transients based on 100 % power condition for uprate case and the differential pressure between the primary and secondary side is calculated for all the transients.

The scale factor is defined as

$$\text{Scale factor}_{\text{(uprate)}} = \frac{\text{DeltaP between the primary side \& secondary side (uprate)}}{\text{DeltaP between the primary side \& secondary side (Stress report)}}$$

Similarly the scale factors for other transients were calculated and these scale factors were applied to scale the primary plus secondary stress range and alternating stresses from the stress report to calculate revised stress range and fatigue usage due to uprate

The following scale factors are calculated for other transients.

Transients	Scale factor
Plant loading, 5 % min	1.051
Plant unloading, 5 % min	1.051
Small step Load Increase	1.046
Small step Load Decrease	1.059
Large step Load Decrease	1.083
Loss of Load	1.067
Loss of power	1.113
Loss of Flow	1.094
Reactor trip from full power	1.112

The sample calculation is shown for 100 % power (Plant Loading)

Primary side pressure = 2250 psi

Secondary side pressure (P_{stm}), 30 % plugging, 1.4 % uprate = 716 psi (PCWG –2579)

Secondary side pressure (P_{stm}) in Stress report analysis = 790 psi

$$\text{Scale factor} = \frac{\text{DeltaP}_{(uprate)}}{\text{DeltaP}_{(Stress\ report)}}$$

$$\text{Scale factor (100 \% power)} = \frac{(2250-716)_{uprate}}{(2250-790)_{Stress\ Report}}$$

$$\text{Scale factor (100 \% power)} = \frac{(1534)_{uprate}}{(1460)_{Stress\ Report}}$$

Scale factor (1.4 % uprate) = 1.051 (@ 100 % power).

Similarly the scale factors are calculated for the other transients.

The evaluation was done based on the Model 51 Stress Reports.

Beaver Valley Unit 1: ASME Boiler and Pressure Vessel Code Section III, " 1965 Edition through winter 1966 Addendum.

Beaver Valley Unit 2 : ASME Boiler and Pressure Vessel Code Section III, " Rules for the construction of Nuclear Vessels, " 1971

U bend fatigue : 1.4 % uprate, 30% plugging:

U bend fatigue evaluation was done due to 1.4 %, 30 % plugging.

Due to flow induced vibration for 1.4% uprate, 30% plugging (Ref : LTR-SGDA-00-277, dated October 30, 2000), three tubes (3) were recommended to be plugged since the fatigue usage is greater than 1.0 for 40 year life. The current tubes identified are valid for one cycle from the implementation of uprating.

Unit-1 S/G C –R10C53

Unit 2 S/G-A –R8C60

Unit 2 S/G-C- R8C69

5. In reference to Section 3.8.11, you stated that the piping systems evaluated for power uprate effects included the reactor coolant (including primary loop piping, primary equipment nozzles, primary equipment supports, and auxiliary piping), main steam, feedwater, high-pressure heater drains, CCW, and fuel pool cooling piping systems. The evaluations performed have concluded that these piping systems remain acceptable and will continue to satisfy design basis requirements in accordance with applicable design basis criteria, when considering the temperature, pressure, and flow rate effects resulting from the power uprate conditions. Discuss your basis for the above conclusion. Provide information (i.e., existing minimum margin in stress and CUF) to demonstrate

that the design basis analysis for the NSSS piping systems reflect sufficient margin to accommodate the changes in the RCS temperatures, or provide the stresses and CUFs in terms of allowable for the most critical piping systems.

Response:

The following discussion is provided to support the conclusion that reactor coolant loop and balance of plant (BOP) piping systems will remain acceptable and will continue to satisfy design basis requirements.

Introduction

The purpose of the piping and support review was to evaluate balance of plant (BOP) piping systems and Reactor Coolant Loop (RCL) piping for the effects resulting from thermal power uprated conditions to demonstrate design compliance. The piping systems evaluated for power uprate effects included the reactor coolant, main steam, feedwater, high pressure heater drains, components cooling water, and fuel pool cooling piping systems. Operation at the thermal power uprated conditions may increase piping stresses caused by slightly higher operating temperatures, pressures and flow rates. Additionally, pipe supports and equipment nozzles may be potentially subjected to slightly increased loadings due to the thermal power uprate condition.

Engineering Evaluation

The methodology used in the piping system evaluations involved reviewing existing piping system stress levels to ensure that adequate design margin was available to accommodate the effects resulting from power uprate. No new computer codes were used in the piping system evaluations.

The piping system evaluations performed to determine system acceptability for the power uprated conditions are summarized as follows:

A review of the thermal power uprate data provided in the heat balance diagrams and Westinghouse PCWG parameters identifies extremely small piping temperature increases (less than 3 degrees F) for the main steam and feedwater piping systems. Associated branch piping systems, as such, will also experience only minor temperature increases as a result of power uprate. Minor temperature increases of this magnitude have been concluded to be acceptable since these increases are offset by inherent conservatisms in analytical methods used to calculate the existing thermal stresses and loads. System operating pressure increases are also very small (i.e., approximately 2%) and will have an insignificant effect on overall piping system stress levels. The steam generator flow rate increase associated with power uprate conditions is less than one and one half percent. This minor increase will have an insignificant effect on existing fluid transient analyses and associated piping stress levels and loads.

To further support the evaluation summary described above, a sampling of Unit 1 and 2 pipe stress calculations was performed to obtain existing thermal expansion stress levels from detailed computer analyses and compare these stresses to applicable design basis limits. This sampling included a total of 39 pipe stress calculations (24 for Unit 1 and 15 for Unit 2) and included piping for the main steam, feedwater, component cooling, heater drains, steam generator blowdown, and fuel pool cooling piping systems. A summary of the existing maximum "expansion stress" levels for these stress calculations and applicable allowable stress values are provided in Tables 1 and 2. Based on a review of the stress data summarized in these tables, it can be concluded that the piping systems associated with these calculations will remain within allowable stress limits for the power uprate condition.

The reactor coolant system (RCS) is essentially unchanged for the power uprate condition. A review of the thermal uprate data contained in Westinghouse's PCWG parameters reveals no change in system pressure and a slight decrease in system flowrate. The Westinghouse PCWG parameters does indicate an extremely small piping temperature increase of less than 1 degree F, which will not result in any significant impact (less than one percent thermal expansion changes) to existing RCL piping system qualifications. Specifically, the existing RCL piping system evaluations, and associated results and conclusions, can be considered unchanged for this minor temperature increase. In summary, it is concluded that the RCL system will remain within allowable limits for the power uprate condition.

The evaluation also performed a review to assess the effects of the power uprate on pipe break, jet, and whip. Due to the resulting small increase in pipe stresses, no new postulated pipe break locations were identified in high-energy piping. In addition, since the uprate only results in a small increase in pressure, no significant increase in jet impingement loading or pipe whip forces will be experienced.

Results/Conclusions

The piping systems review concluded that all piping systems remain acceptable and will continue to satisfy design basis requirements in accordance with applicable design basis criteria, when considering the temperature, pressure and flow rate effects resulting from the power uprate conditions. Specifically, Beaver Valley Unit 1 piping and related support systems remain within allowable stress limits in accordance with American National Standards Institute (ANSI) B31.1, 1967 edition, including the 1971 Addenda. Beaver Valley Unit 2 piping and related support systems remain within allowable stress limits in accordance with ASME Section III, 1971 edition, including Addenda through the Winter 1972 for Class 1, 2, and 3 piping, and ANSI B31.1, 1967 edition, including Addenda through June 30, 1972 for Class 4 piping. The evaluations also document that no piping or pipe support modifications are required as a result of the increased power level.

Table 1 Beaver Valley Unit 1 Sample Stress Calculations

SYSTEM		TEMP (°F)	PRESS (PSIG)	MAXIMUM EXPANSION STRESS (PSI)	EXPAN. STRESS ALLOWABLE (PSI)
CCP		152	80	19465	22500
CCP		152	110	8179	22500
CCP		152	20	4747	22500
CCP		152	110	4499	22500
CCP		152	110	4252	22500
CCP		152	20	14070	22500
CCP		152	110	7661	22500
CCP		124	110	6434	22500
CCP		124	110	8115	22500
CCP		124	110	2655	22500
CCP		124	110	1106	22500
CCP		152	110	3969	22500
CCP		124	110	7727	22500
CCP		152	110	33334 (Note 1)	34172 (Note 2)
CCP		124	110	3516	22500
CCP		152	110	21070	22500
CCP		124	110	15096	22500
CCP		152	110	10113	22500
CCP		70	75	6414	22500
FEEDWATER		441	899	18674	22500
FEEDWATER		441	899	13855	30000
FEEDWATER		441	899	22481	22500
FEEDWATER		441	899	15505	30000
FEEDWATER		441	899	29225 (Note 1)	37500 (Note 2)
FEEDWATER		441	899	15666	30000
FUEL POOL COOL		148	45	4444	27861
FUEL POOL COOL		148	13	5664	27861
RCL (LOOP A)		653	2485	6836	18785
RCL (LOOP B)		653	2485	8134	18785
RCL (LOOP C)		653	2485	7590	18785

Notes:
1. Maximum Sustained + Expansion Stress
2. Sustained + Expansion Stress Allowable

Table 2 Beaver Valley Unit 2 Sample Stress Calculations

SYSTEM	TEMP (°F)	PRESS (PSIG)	MAXIMUM EXPANSION STRESS (PSI)	EXPAN. STRESS ALLOWABLE (PSI)
FEED WATER	450	1058	8216	18000
FEED WATER	450	1058	26455 (Note 1)	33000 (Note 2)
FEED WATER	450	1600	31838 (Note 1)	37500 (Note 2)
FEED WATER	450	1085	32673 (Note 1)	33000 (Note 2)
FEED WATER	450	1600	13945	18000
H PRESS HEATER DRAIN	395	215	1790	22500
MAIN STEAM	547	1085	24662	26250
MAIN STEAM	547	1085	22481	26250
MAIN STEAM	547	1085	18381	26250
MAIN STEAM	560	1085	18675	22500
MAIN STEAM	547	1005	19427	22500
MAIN STEAM	560	1085	32836 (Note 1)	37500 (Note 2)
MAIN STEAM	560	1085	20814	22500
MAIN STEAM	560	1085	33581 (Note 1)	37500 (Note 2)
MAIN STEAM	560	1085	26150 (Note 1)	37500 (Note 2)
RCL (LOOP A) Note 3	650	2485	25876	48300
RCL (LOOP B) Note 4	650	2485	24389	48300
RCL (LOOP C) Note 5	650	2485	36638	48300

Notes:
1. Maximum Sustained + Expansion Stress
2. Sustained + Expansion Stress Allowable
3. Maximum Cumulative Usage Factor for RCL Loop A is equal to 0.9972
4. Maximum Cumulative Usage Factor for RCL Loop B is equal to 0.9672
5. Maximum Cumulative Usage Factor for RCL Loop C is equal to 0.9681

REFERENCE

FirstEnergy Nuclear Operating Company Letter to the NRC, "Request For Additional Information, License Power Uprate Amendment Request Nos 289 And 161," dated January 18, 2001, Enclosure 1, "Beaver Valley Power Station, Units 1 And 2, 1.4 Percent Power Uprate Program, FENOC Licensing Submittal."