FENOC
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June 29, 2001
L-01-084

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) pertaining to FENOC letter L-01-078, dated June 9, 2001. FENOC letter L-01-078 provided responses to an earlier RAI in support of License Amendment Requests (LAR) 289 and 161. These LARs were submitted by FENOC letter L-01-006 dated January 18, 2001, and proposed a 1.4% power uprate for both Beaver Valley Power Station (BVPS) units. This letter also contains the withdrawal of a Bases change proposed in FENOC letter L-00-143. This letter, which transmitted LARs 286 and 158, proposed the utilization of the Revised Thermal Design Procedure (RTDP) methodology and relocated certain Technical Specification requirements to the Licensing Requirements Manual (LRM). Also contained in this letter is the FENOC response to a RAI received on June 12, 2001, pertaining to the loss of coolant accident analyses parameters that affect the peak clad temperature analysis.

The information provided by this letter consists of the following:

- elaboration of the response provided to request number 2a of letter L-01-078,
- revision to Table A-1 of letter L-01-078,
- response to the peak clad temperature analysis RAI received on June 12, 2001, and
- withdrawal of a proposed Bases change contained in FENOC letter L-00-143.

The FENOC responses are provided in Attachment A of this letter. Revised marked up Bases pages reflecting the withdrawal of a change proposed in FENOC letter L-00-143 are provided in Attachment B of this letter.

A001

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 for LARs 289 and 161 is requested following the effective date of these amendments.

This information does not change the evaluations or conclusions presented in FENOC letter L-00-143 or 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,

Marc Pearson for

Lew W. Myers

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, Marc P. Pearson, being duly sworn, state that I am Director, Nuclear Services 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

Marc P. Pearson

Marc P. Pearson
Director, Nuclear Services - 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 29 th day of June, 2001.

Tracey A. Baczek

My Commission Expires:

Notarial Seal
Tracey A. Baczek, Notary Public
Shippingport Boro, Beaver County
My Commission Expires Aug. 16, 2001
Member, Pennsylvania Association of Notaries

Letter L-01-084 - Attachment A

NRC Request pertaining to the response to request number 2a of L-01-078

The BV response to request number 2a states, "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."

NRC Concern

The power uprate causes certain changes in primary and secondary side parameters during full power operation. Because of these parameter changes, the initial plant conditions assumed for the accident analyses should not be "identical" to the previous analyses (e.g. primary and secondary pressure, temperature, water level, etc.). Please provide the initial plant conditions used for the accident analyses and discuss how they affect the results of each event analyzed.

FENOC Response

The non loss of coolant accident (LOCA) analyses performed in support of the Revised Thermal Design Procedure (RTDP) methodology and 1.4% uprating programs were explicitly performed at conditions consistent with the uprated power level. It should be pointed out that nominal initial conditions are assumed consistent with RTDP methodology (see WCAP-11397-P-A / WCAP-11397-A). The specific initial conditions are as follows:

NSSS	2697 MWt
Core power	2689 MWt
Full Power Vessel T_{avg}	576.2°F
No Load Temperature	547 °F
Primary Pressure	2250 psia
Secondary Pressure	806 psia (0% S/G Tube Plugging)
Secondary Pressure	716 psia (30% S/G Tube Plugging)
Minimum measured RCS flow	266,800 gpm

Of these parameters, only the Nuclear Steam Supply System (NSSS) Power, Core Power, and Secondary Pressure have changed from the pre-uprate nominal conditions. The effects of the change in secondary pressure (approximately 5 psia decrease) is insignificant on the results of the accidents and the effect of the increased power (1.4 %) is to reduce Departure from Nucleate Boiling Ratio (DNBR) margin by approximately this amount.

NRC Request pertaining to Table A-1 of L-01-078

Table A-1 of the BV response shows the minimum DNBR for the Complete Loss of Flow event to be below the DNBR limits for both plants. A footnote states that DNB Margin has been allocated for these cases.

NRC Concern

A Complete Loss of Flow is classified as an incident with moderate frequency, which is not allowed to have fuel failure. Please explain why it is acceptable for the DNBR in this event to be below the minimum DNBR established for BV (1.36). Also, please explain what is meant by the footnote, "DNB Margin has been allocated for these cases," and explain how this relates to the adequacy of the analyses' DNBR ratio.

FENOC Response

The DNBR limit is the 95/95 design limit of 1.24 (typical) and 1.23 (thimble). The safety analyses limit (SAL) was set to 1.36 in order to preserve some DNBR margin for future use and to accommodate a generic rod bow penalty of 1.3%. With the 1.36 SAL, approximately 9% margin was maintained for future use and rod bow. Out of this 9% margin, an allocation of 3.3% was assigned for the 1.4% uprate. This would have effectively lowered the SAL limit to 1.29. The results of the revised safety analyses show a minimum DNBR of 1.335 and 1.335 for the Complete Loss of Flow event for Unit 1 and Unit 2, respectively. Therefore, to bound the current analyses at uprate conditions, the SAL can be set at 1.33. With the revised SAL of 1.33, the following DNB margin exists:

	Typical	Thimble
DNBR Safety Analyses Limit	1.33	1.33
DNBR Design Limit	1.24	1.23
DNBR Margin	6.8%	7.5%
Rod Bow Penalty	1.3%	1.3%
Remaining Margin	5.5%	6.2%

The attached Table A-1, which replaces that provided in L-01-078, reflects the actual analysis minimum DNBR for all events listed and the revised SAL DNBR of 1.33. The results showing the peak primary and secondary pressure for BVPS Unit 1 for the Loss of Load event have been revised. These changes reflect the results of a revised analysis, which was performed to correct an input error. The error did not affect the conclusions of the analysis. The results showing the minimum DNBR and peak primary pressure for BVPS Unit 2 for the Complete Loss of Flow event have also been revised to correct typographical errors. These typographical errors did not affect the conclusions of the analysis. The changes made to Table A-1 are denoted with revision bars.

The revision to a SAL DNBR of 1.33 requires that a Bases change proposed by LARs 286 and 158 be withdrawn. The change being withdrawn would have changed the

DNBR limit appearing on page B 2-1 for each Beaver Valley Power Station unit to 1.36. Withdrawing this change restores the limit to its present value of 1.33. The revised marked up Bases pages are provided in Attachment B of this letter.

NRC June 12, 2001 Request

To show that the Westinghouse generically approved LBLOCA and SBLOCA analysis methodologies continue to apply specifically to the Beaver Valley plant(s) provide a statement, if appropriate, that Beaver Valley and its vendor have ongoing processes which assure that LOCA analysis input values for peak cladding temperature-sensitive parameters bound the as-operated plant values for those parameters.

FENOC Response

FENOC and its vendor have ongoing processes which assure that LOCA analysis input values for peak cladding temperature-sensitive parameters bound the as-operated plant values for those parameters. Examples include Technical Specifications and the Licensing Requirements Manual which provide limits and surveillance requirements for most of the sensitive parameters including, by reference to ASME Section XI requirements, performance requirements for the emergency core cooling system components. The Reload Safety Evaluation Process (WCAP 9272-P-A) provides the process for ensuring that the core design complies with assumptions in the current large break LOCA and small break LOCA safety analyses.

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) Rod Withdrawal at Power (Unit 2)	14.1.2 15.4.2	1.370 1.362	N/A * N/A *	1171 psia 1171 psia
Partial Loss of Flow (Unit 1) Partial Loss of Flow (Unit 2)	14.1.5 15.3.1	1.787 1.790	2339.5 psia 2327.8 psia	922.2 psia 920.6 psia
Loss of Load - DNB Case (Unit 1) Loss of Load - DNB Case (Unit 2)	14.1.7 15.2.2/15.2.3	1.72 1.67	2675.2 psia 2747.5 psia	1177.4 psia 1182.5 psia
Feedwater Malfunction (Unit 1) Feedwater Malfunction (Unit 2)	14.1.9 15.1.1/15.1.2	1.835 1.894	2338 psia 2341 psia	1123 psia 1179 psia
Excessive Load Increase (Unit 1) Excessive Load Increase (Unit 2)	14.1.10 15.1.3	>1.33 >1.33	N/A N/A	N/A N/A
RCS Depressurization (Unit 1) RCS Depressurization (Unit 2)	14.1.15 15.6.1	1.65 1.76	N/A N/A	N/A N/A
Complete Loss of Flow (Unit 1) Complete Loss of Flow (Unit 2)	14.2.9 15.3.2	1.335 1.335	2421.1 psia 2114.2 psia	949.4 psia 951.0 psia
Limits	---	1.33	2748.5 psia	1208.5 psia

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

Letter L-01-084 - Attachment B

Replacement Pages for marked up pages contained in LARs 286 and 158

LAR	Unit	Page
286	1	B 2-1
158	2	B 2-1

Revised Markup for Unit 1 LAR 286

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the WRB-1 correlation. The WRB-1 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. REPLACE WITH INSERT "4"

~~The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting fuel rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 correlation in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit (1.17 for the WRB-1 correlation).~~

REPLACE WITH INSERT "5"

~~In meeting this design basis, uncertainties in nuclear and thermal parameters, and fuel fabrication parameters were combined statistically with the DNB correlation uncertainties to determine the plant DNBR uncertainty and establish the design DNBR limit such that there is at least a 95% probability with 95% confidence level that the minimum DNBR for the limiting fuel rod is greater than or equal to the DNBR limit. For this application, the design DNBR limit is 1.21. This DNBR value must be met in plant safety analyses using nominal values of the input parameters that were included in the DNBR uncertainty evaluation.~~ In addition, margin has been maintained in the design by meeting a safety analysis DNBR limit of 1.33 in performing safety analyses.

The ~~curves of Figure 2.1-1. show~~ figure provided in the COLR shows the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the safety analysis DNBR limit or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

ADD INSERT "6"

(Proposed Wording)

Revised Markup for Unit 2 LAR 158

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the WRB-1 correlation. The WRB-1 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

REPLACE WITH INSERT "4"

~~The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting fuel rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 correlation in this application). The correlation DNBR limit is based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit (1.17 for the WRB-1 correlation).~~

REPLACE WITH INSERT "5"

~~Incorporating the peaking factor uncertainties in the correlation limit results in a DNBR design limit value of 1.21. This DNBR value must be met in plant safety analyses using nominal values of the input parameters that were included in the DNBR uncertainty evaluation.~~ In addition, margin has been maintained in the design by meeting a safety analysis DNBR limit of 1.33 in performing safety analyses.

The ~~curve of Figure 2.1-1~~ figure provided in the COLR shows the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the safety analysis DNBR limit, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

~~The curves are figure is based on enthalpy hot channel factor limits provided in the Core Operating Limits Report (COLR).~~

DELETE

(Proposed Wording)