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March 11, 2003
L-03-020

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
License Amendment Request Nos. 303 and 174

Pursuant to 10 CFR 50.90, FirstEnergy Nuclear Operating Company (FENOC) requests an amendment to the above licenses in the form of changes to the technical specifications. The proposed change requests approval to apply the Westinghouse best-estimate large break loss of coolant accident analysis methodology to Beaver Valley Power Station Units 1 and 2, and requests amendment of the respective technical specifications. This best-estimate methodology has previously been approved on a generic basis by the NRC.

This license amendment request (LAR) contains one enclosure with four attachments. The proposed technical specification changes are provided in Attachments A-1 and A-2 for Units 1 and 2, respectively. The changes to technical specification bases are provided in Attachments B-1 and B-2 for Units 1 and 2, respectively. Attachment C describes commitments contained in this submittal.

A best-estimate loss of coolant accident analysis has been completed for each unit assuming an atmospheric containment. Therefore, approval of this LAR for each unit is contingent upon approval of the containment conversion LAR for the corresponding unit (i.e., LARs 300 and 172 for Units 1 and 2, submitted by letter L-02-069 dated June 5, 2002). Thus, the implementation dates for the best estimate license amendments should be consistent with the implementation dates of the corresponding containment conversion amendment.

Therefore, for each unit's best estimate amendment, FENOC is requesting an implementation period of 60 days following implementation of its containment conversion amendment.

A001

The Beaver Valley review committees have reviewed this change. The change was determined to be safe and does not involve a significant hazard consideration as defined in 10 CFR 50.92 based on the attached safety analysis and no significant hazard evaluation.

If there are any questions concerning this matter, please contact Mr. Larry R. Freeland, Manager, Regulatory Affairs/Performance Improvement at 724-682-5284.

I declare under penalty of perjury that the foregoing is true and correct. Executed on March 11, 2003.

Sincerely,

Mark B. Bezilla

Enclosure: License Amendment Requests Nos. 303 (Unit 1) and 174 (Unit 2)

Attachments: A-1, BVPS - Unit 1 Technical Specification Changes
A-2, BVPS - Unit 2 Technical Specification Changes
B-1, BVPS - Unit 1 Technical Specification Bases Changes
B-2, BVPS - Unit 2 Technical Specification Bases Changes
C, Commitments

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

ENCLOSURE 1

Beaver Valley Power Station, Unit Nos. 1 and 2
License Amendment Requests Nos. 303 (Unit 1) and 174 (Unit 2)

FirstEnergy Nuclear Operating Company Evaluation

Subject: Application to Permit Operation with Best-Estimate Large Break LOCA Methodology.

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A-2	Proposed Unit 2 Technical Specification Changes
B-1	Proposed Unit 1 Technical Specification Bases Changes
B-2	Proposed Unit 2 Technical Specification Bases Changes
C	Commitments

1.0 DESCRIPTION

This license amendment request (LAR) for operating licenses DPR-66 (Beaver Valley Power Station Unit 1) and NPF-73 (Beaver Valley Power Station Unit 2) requests approval to apply the Westinghouse best-estimate large break loss of coolant accident (LOCA) analysis methodology. It is requested that Technical Specification 6.9.5, "Core Operating Limits Report (COLR)" be amended to allow use of the methodology. The specific changes to the technical specifications (TS) that are proposed are shown on Attachments A-1 and A-2 for Beaver Valley Power Station (BVPS) Units 1 and 2, respectively. Changes to the respective TS Bases are submitted for information in Attachments B-1 and B-2. Attachment C describes commitments contained in this submittal.

2.0 PROPOSED CHANGES

TS 6.9.5.b lists applicable references for the analytical methods used to determine core operating limits identified in TS 6.9.5.a. This list of references includes the Westinghouse topical report that documents the currently approved large break LOCA analysis methodology. It is proposed that this reference would be replaced with the generically approved topical report for the Westinghouse best-estimate large break LOCA analysis methodology (WCAP-12945-P-A).

3.0 BACKGROUND

Westinghouse has obtained generic NRC approval of its topical report describing best-estimate large break LOCA methodology. NRC approval of the methodology is documented in the NRC safety evaluation report appended to the topical report (WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best Estimate LOCA Analysis," March 1998). Separate plant specific analyses for BVPS Units 1 and 2 have been performed using the approved methodology.

These changes are being made to incorporate the best-estimate approach into the licensing basis for BVPS large break LOCA analyses in accordance with 10 CFR 50.46, Regulatory Guide 1.157 "Best-Estimate Calculations of Emergency Core Cooling System Performance," and the Westinghouse "Code Qualification Document For Best Estimate LOCA Analysis," WCAP-12945-P-A, Volumes 1-5. Best-estimate methodology is needed to support a future extended power uprate of the BVPS units and its use is dependent on implementation of atmospheric containment conversion (License Amendment Requests (LARs) 300 (Unit 1) and 172 (Unit 2), submitted separately by FENOC letter L-02-069 dated June 5, 2002). Completed best-estimate LOCA analyses have been performed at the planned uprated conditions (2900 MWt) with an atmospheric containment. The values of major plant parameters assumed in the best-estimate LOCA analyses will be documented in the respective Updated Final Safety Analysis Report (UFSAR) for each unit. These and other UFSAR changes resulting from approval of this LAR will be made in accordance with 50.71(e).

Both FirstEnergy Nuclear Operating Company (FENOC) and its analysis vendor (Westinghouse) have ongoing processes in place that assure that analysis input values for peak clad temperature-sensitive parameters bound their as-operated plant values.

4.0 TECHNICAL ANALYSIS

Separate best-estimate large break loss of coolant accident analyses have been performed for BVPS Units 1 and 2 using the methodology contained in WCAP-12945-P-A. All plant specific parameters used in the analyses are bounded by the models and correlations contained in the generic methodology. Therefore, the BVPS analyses conform to 10 CFR 50.46 and Section II of Appendix K, and meet the intent of Regulatory Guide 1.157. The conclusions of the analyses are that there is a high level of probability that:

1. The calculated maximum fuel element cladding temperature (peak cladding temperature) will not exceed 2200°F.

2. The calculated total oxidation of the cladding (maximum cladding oxidation) will not exceed 0.17 times the total cladding thickness before oxidation.

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam (maximum hydrogen generation) will nowhere exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4. The calculated changes in core geometry are such that the core remains amenable to cooling.

5. After successful initial operation of the ECCS, the core temperature will be maintained at an acceptably low value and decay heat will be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

Tables 1 and 2 present the 95th percentile peak clad temperature (PCT), maximum cladding oxidation, maximum hydrogen generation, and cooling results for BVPS Units 1 and 2, respectively.

Therefore, FENOC has concluded that adopting the best-estimate large break LOCA methodology for BVPS Units 1 and 2 and making the proposed TS changes would not adversely affect the health and safety of the public.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

FirstEnergy Nuclear Operating Company has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   No. No physical changes are required as a result of implementing best-estimate large break loss of coolant accident (LOCA) methodology and associated technical specification changes. The plant conditions assumed in the analysis are bounded by the design conditions for all equipment in the plant. Therefore, there will be no increase in the probability of a loss of coolant accident. The consequences of a LOCA are not being increased, since it is shown that the emergency core cooling system is designed so that its calculated cooling performance conforms to the criteria contained in 10 CFR 50.46, Paragraph b. No other accident is potentially affected by this change.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any previously analyzed?

   No. There are no physical changes being made to the plants. No new modes of plant operation are being introduced. The parameters assumed in the analysis are within the design limits of the existing plant equipment. All plant systems will perform as designed during the response to a potential accident.

   Therefore, the proposed change does not involve an increase in the probability or consequences of an accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in the margin of safety?

   No. It has been shown that the methodology used in the analysis would more realistically describe the expected behavior of plant systems during a postulated loss of coolant accident. Uncertainties have been accounted for as required by 10 CFR 50.46. A sufficient number of loss of coolant accidents with different break sizes, different locations and other variations in properties are analyzed to provide assurance that the most severe postulated loss of coolant accidents are calculated. It has been shown by analysis that there is a high level of probability that all criteria contained in 10 CFR 50.46, Paragraph b are met.

5.2 Applicable Regulatory Requirements/Criteria

The proposed changes have been evaluated to determine whether applicable regulations and requirements would continue to be met. FENOC has determined that the proposed changes do not require any exemptions or relief from regulatory requirements, other than the TS, and do not affect conformance with any GDC differently than described in the SAR. Section 4 of this analysis demonstrates that the proposed change is consistent with 10 CFR 50.46.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

Based on this evaluation and the fact that either an environmental impact statement or an environmental assessment is required, the proposed amendment will not have an adverse effect on the environment and can thus be deemed acceptable.

7.0 REFERENCES

1. WCAP 12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best Estimate LOCA Analysis," March 1998

2. Regulatory Guide 1.157 "Best-Estimate Calculations of Emergency Core Cooling System Performance (Draft RS 701-4 published 3/1987)."

3. NUREG-0800, Standard Review Plan, "Emergency Core Cooling"

4. 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

5. 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors."

6. 10 CFR 50.71(e), "Maintenance of records, making of reports."

Table 1

BEAVER VALLEY UNIT 1 BEST-ESTIMATE
LARGE BREAK LOCA RESULTS

	Value	**Acceptance Criteria**
95th Percentile PCT (°F)*	2144**	2200
Maximum Cladding Oxidation (%)*	10.3	17
Maximum Hydrogen Generation (%)*	0.92	1
Coolable Geometry	Core Remains Coolable	Core Remains Coolable
Long Term Cooling	Core Remains Cool in Long Term	Core Remains Cool in Long Term

* Calculated using the methodology in the following reference:

WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best-Estimate Loss-of-Coolant Accident Analysis," March 1998 (Westinghouse Proprietary).

** The final licensing basis result including all evaluations is 2158°F [2144°F (MONTECF 95^{th} percentile PCT) + 14°F (Mixed Core Penalty)]

Table 2

BEAVER VALLEY UNIT 2 BEST-ESTIMATE
LARGE BREAK LOCA RESULTS

	Value	**Acceptance Criteria**
95th Percentile PCT (°F)*	1976**	2200
Maximum Cladding Oxidation (%)*	6.7	17
Maximum Hydrogen Generation (%)*	0.89	1
Coolable Geometry	Core Remains Coolable	Core Remains Coolable
Long Term Cooling	Core Remains Cool in Long Term	Core Remains Cool in Long Term

* Calculated using the methodology in the following reference:

WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best-Estimate Loss-of-Coolant Accident Analysis," March 1998 (Westinghouse Proprietary).

** The final licensing basis result including all evaluations is 1991°F [1976°F (MONTECF 95th percentile PCT) + 15°F (Mixed Core Penalty)]

Attachment A-1

Beaver Valley Power Station Unit No. 1
License Amendment Request No. 303

Proposed Technical Specification Changes (mark-ups)

The following is the affected page:

6-19

CORE OPERATING LIMITS REPORT (Continued)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (Westinghouse Proprietary).

WCAP-8745-P-A, Design Bases for the Thermal Overtemperature ΔT and Thermal Overpower ΔT trip functions, September 1986.

~~WCAP-10266-P-A Rev. 2/WCAP-11524-NP-A Rev. 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," Kabadi, J. N., March 1987; including Addendum 1-A "Power Shape Sensitivity Studies" 12/87 and Addendum 2-A "BASH Methodology Improvements and Reliability Enhancements" 5/88.~~ WCAP 12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best Estimate LOCA Analysis," March 1998 (Westinghouse Proprietary).

WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT." September 1974 (Westinghouse Proprietary).

T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980 -- Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.

NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981.

WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995 (Westinghouse Proprietary).

As described in reference documents listed above, when an initial assumed power level of 102% of rated thermal power is specified in a previously approved method, 100.6% of rated thermal power may be used when input for reactor thermal power measurement of feedwater flow is by the leading edge flow meter (LEFM).

Caldon, Inc. Engineering Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√™ System," Revision 0, March 1997.

Attachment A-2

Beaver Valley Power Station Unit No. 2
License Amendment Request No. 174

Proposed Technical Specification Changes (mark-ups)

The following is the affected page:

6-20

REPORTING REQUIREMENTS (Continued)

WCAP-8745-P-A, "Design Bases for the Thermal Overtemperature ΔT and Thermal Overpower ΔT Trip Functions," September 1986.

~~WCAP-10266-P-A Rev. 2/WCAP-11524-NP-A Rev. 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," Kabadi, J. N., March 1987; including Addendum 1-A "Power Shape Sensitivity Studies" 12/87 and Addendum 2-A "BASH Methodology Improvements and Reliability Enhancements" 5/88.~~ WCAP 12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best Estimate LOCA Analysis," March 1998 (Westinghouse Proprietary).

WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT." September 1974 (Westinghouse Proprietary).

T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980 -- Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.

NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981.

WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995 (Westinghouse Proprietary).

As described in reference documents listed above, when an initial assumed power level of 102% of rated thermal power is specified in a previously approved method, 100.6% of rated thermal power may be used when input for reactor thermal power measurement of feedwater flow is by the leading edge flow meter (LEFM).

Caldon, Inc. Engineering Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√™ System," Revision 0, March 1997.

Caldon, Inc. Engineering Report-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM√™ System," Revision 0, May 2000.

Attachment B-1

Beaver Valley Power Station Unit No. 1
License Amendment Request No. 303

Proposed Technical Specification Basis Change

For Information Only

The following is a list of the affected pages:

B 3/4 2-1
B 3/4 2-4
B 3/4 2-6

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core ≥ the design DNBR limit during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F as specified in 10 CFR 50.46 is not exceeded.

The definitions of hot channel factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are

3/4.2.2 AND 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL FACTORS-$F_Q(Z)$ and $F^N_{\Delta H}$

The limits on heat flux and nuclear enthalpy hot channel factors ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the ECCS acceptance criteria limit of 2200°F as specified in 10 CFR 50.46.

Each of these hot channel factors are measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and Specification 4.2.3. This periodic surveillance is sufficient to insure that the hot channel factor limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ±12 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.5.

c. The control rod insertion limits of Specifications 3.1.3.4 and 3.1.3.5 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE is maintained within the limits.

The relaxation in $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F^N_{\Delta H}$ will be maintained within its limits provided conditions a through d above, are maintained.

When a F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. 5% is the appropriate experimental error allowance for a full core map taken with the incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

The specified limit of $F^N_{\Delta H}$ contains an 8% allowance for uncertainties which means that normal, full power, three loop operation will result in $F^N_{\Delta H} \leq$ the design limit specified in the CORE OPERATING LIMITS REPORT.

3/4.2.4 QUADRANT POWER TILT RATIO (QPTR) (Continued)

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident, the peak cladding temperature must not exceed 2200°F in accordance with 10 CFR 50.46 as specified in 10 CFR 50.46;

b. During a loss of forced reactor coolant flow accident, there must be at least 95 percent probability at the 95 percent confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/gm in accordance with the indicated failure threshold from the TREAT results (UFSAR 14.2.6), and

d. The control rods must be capable of shutting down the reactor with a minimum required Shutdown Margin (SDM) with the highest worth control rod stuck fully withdrawn in accordance with 10 CFR 50, Appendix A, GDC 26.

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

The QPTR limits ensure that $F^N_{\Delta H}$ and $F_Q(Z)$ remain below their limiting values by preventing an undetected change in the gross radial power distribution.

In MODE 1, the $F^N_{\Delta H}$ and $F_Q(Z)$ limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analysis.

Attachment B-2

Beaver Valley Power Station Unit No. 2
License Amendment Request No. 174

Proposed Technical Specification Basis Change

For Information Only

The following is a list of the affected pages:

B3/4 2-1
B3/4 2-2
B3/4 2-5

Provided for Information Only.

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core ≥ the design DNBR limit during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded as specified in 10 CFR 50.46.

The definitions of hot channel factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the integral of linear power along the rod with the highest integrated power to the average rod power.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the target band about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time

AXIAL FLUX DIFFERENCE (AFD) (Continued)

duration limit of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits specified in the CORE OPERATING LIMITS REPORT for THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 90% and between 15% and 50% of RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band near the beginning of core life.

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL FACTORS $F_Q(Z)$ and $F^N_{\Delta H}$

The limits on heat flux and nuclear enthalpy hot channel factors ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the ECCS acceptance criteria limit of 2200°F as specified in 10 CFR 50.46.

Each of these hot channel factors are measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the hot channel factor limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6.

3/4.2.4 QUADRANT POWER TILT RATIO (QPTR)

BACKGROUND

The Quadrant Power Tilt Ratio limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation. The QPTR is routinely determined using the power range channel input which is part of the power range nuclear instrumentation (NI). The power range channel provides a protection function and has operability requirements in LCO 3.3.1. While part of the NI channel, the power range channel input to QPTR functions independently of the power range channel in monitoring radial power distribution. For this reason, if the power range channel output is inoperable, the power range channel input to QPTR may be unaffected and capable of monitoring for the QPTR.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.1, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.3.6, "Control Rod Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident, the peak cladding temperature must not exceed 2200°F in accordance with 10 CFR 50.46 as specified in 10 CFR 50.46;

b. During a loss of forced reactor coolant flow accident, there must be at least 95 percent probability at the 95 percent confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/gm in accordance with the indicated failure threshold from the TREAT results (UFSAR 15.4.8), and

Attachment C

Beaver Valley Power Station Unit Nos. 1 and 2
License Amendment Request Nos. 303 and 174

Commitment List

The following list identifies those actions committed to by FirstEnergy Nuclear Operating Company (FENOC) for Beaver Valley Power Station (BVPS) Unit Nos. 1 and 2 in this document. Any other actions discussed in the submittal represent intended or planned actions by Beaver Valley. These other actions are described only as information and are not regulatory commitments. Please notify Mr. Larry R. Freeland, Manager, Regulatory Affairs/Performance Improvement, at Beaver Valley on (724) 682-5284 of any questions regarding this document or associated regulatory commitments.

Commitment	Due Date
The values of major plant parameters assumed in the best-estimate LOCA analyses will be documented in the respective Updated Final Safety Analysis Report (UFSAR) for each unit.	Next scheduled UFSAR update in accordance with 10 CFR 50.71(e) following implementation of the best-estimate LOCA amendment at each unit.